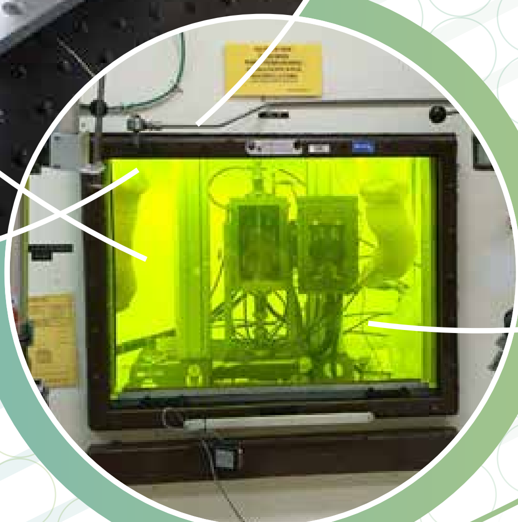
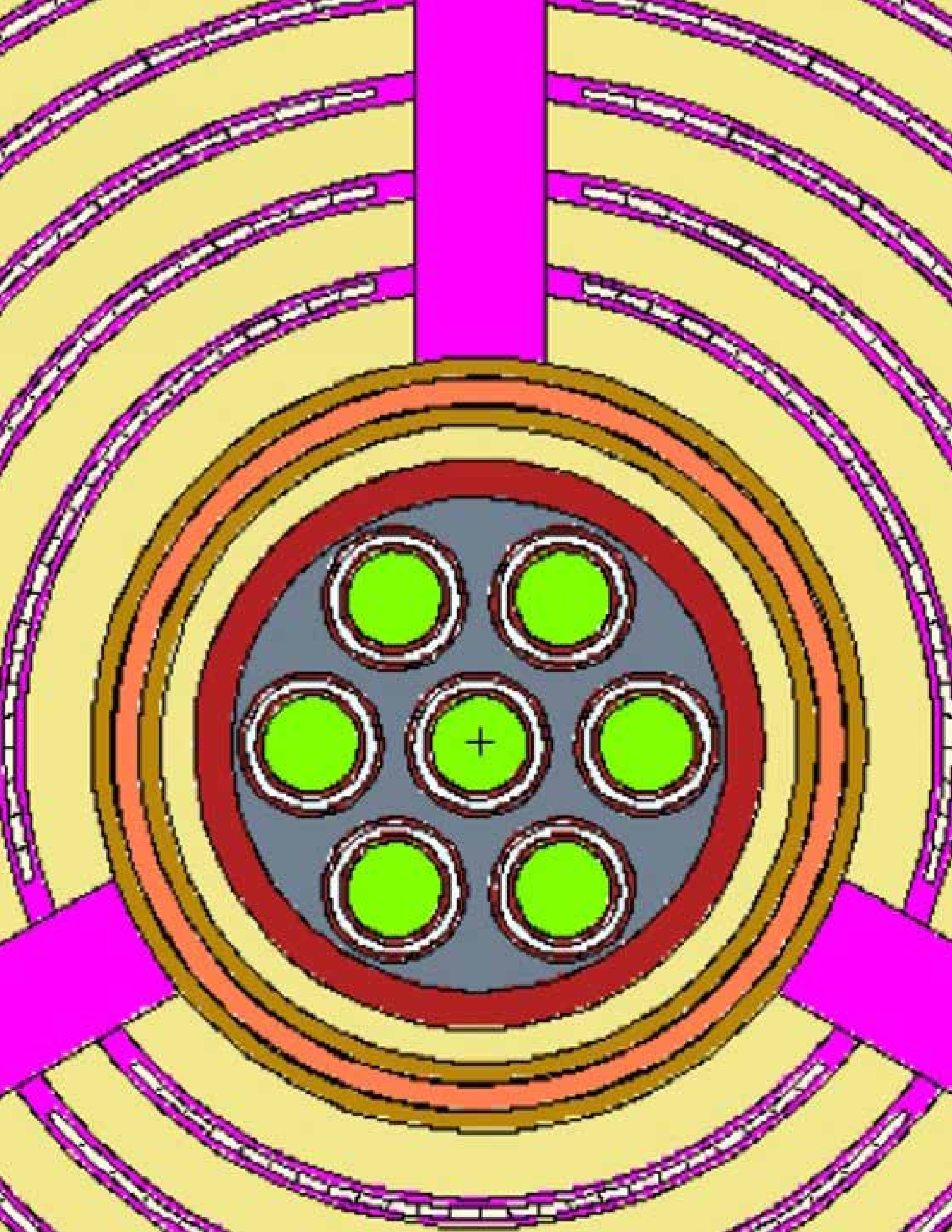


ADVANCED FUELS CAMPAIGN
2020 Accomplishments

AFC Advanced Fuels Campaign



Nuclear Fuels Cycle & Supply Chain

Advanced Fuels Campaign 2020 Accomplishments

INL/EXT-20-60312

November 2020

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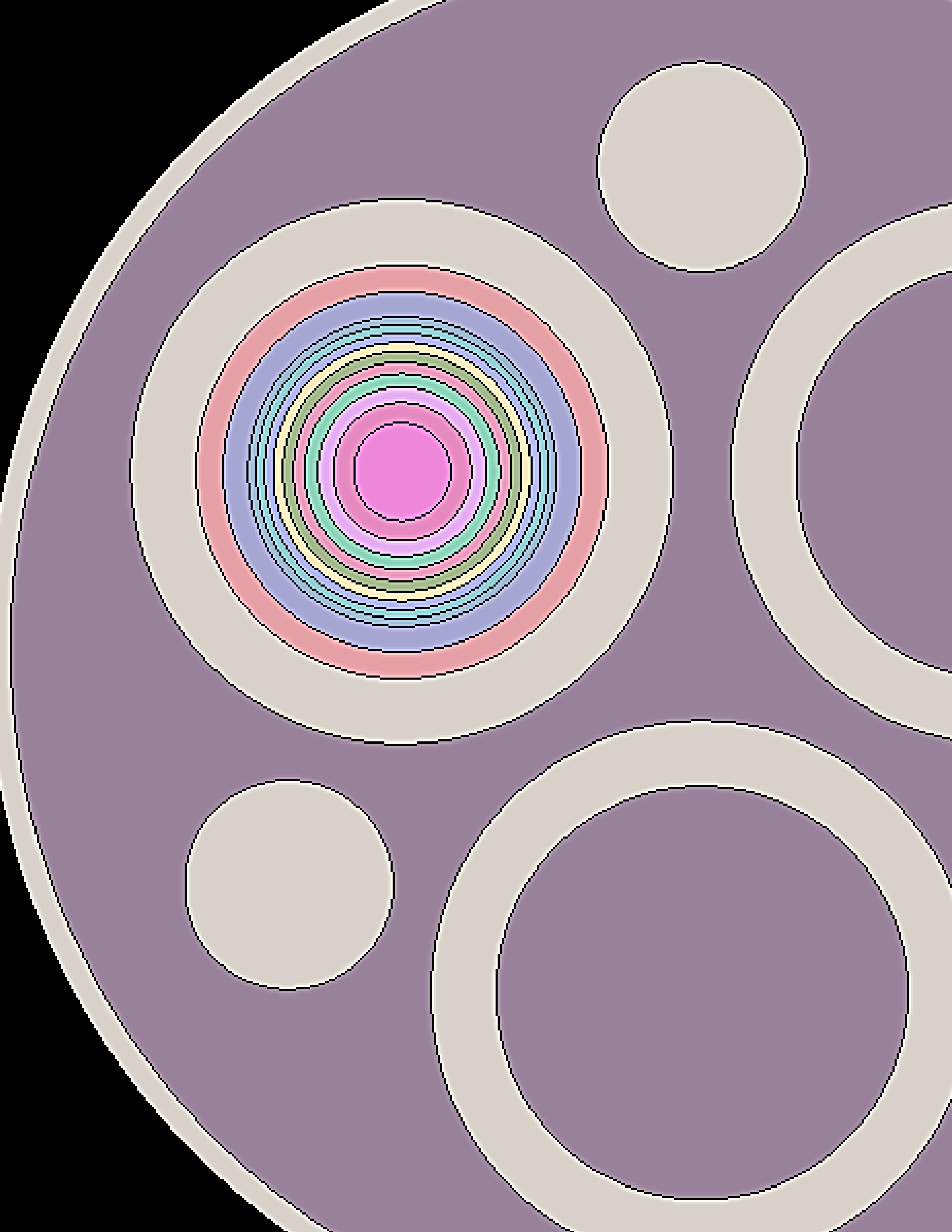
AFC INTEGRATION AND MANAGEMENT



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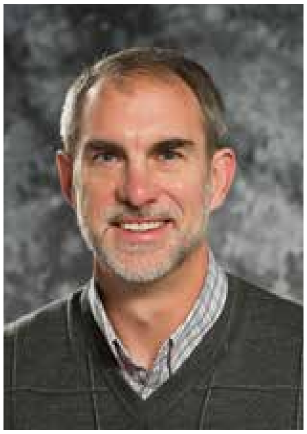

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The mission of the Advanced Fuels Campaign (AFC) is to perform or support research, development, and demonstration (RD&D) activities to identify and mature innovative fuels, cladding materials, and associated technologies with the potential to improve the performance and enhance the safety of current and future reactors; increase the efficient utilization of nuclear energy resources; contribute to enhancing proliferation resistance of the nuclear fuel cycle; and address challenges related to waste management and ultimate disposal.

AFC pursues its mission objectives using a goal-oriented, science-based approach that seeks to establish a fundamental understanding of fuel and cladding behaviors under conditions that arise during fabrication, normal steady-state irradiation, off-normal transient scenarios, and storage/disposal. This approach includes advancing the theoretical understanding of fuel behavior, conducting fundamental and integral experiments, and supporting the mechanistic, multi-scale modeling of nuclear fuels to inform and guide fuel development projects, advance the technological readiness of promising fuel candidates, and ultimately support fuel qualification and licensing initiatives. In the area of advanced tools for the modeling and simulation of nuclear fuels, the AFC works in close partnership with the Nuclear Energy Advanced Modeling and Simulation (NEAMS) program, collaborating with NEAMS in developing mechanistic fuel behavior models and providing experimental data to inform and validate its most advanced tools.

Specifically, AFC objectives in the coming five year horizon include:

1. support the industry-led development of Accident Tolerant Fuel (ATF) technologies with improved reliability and performance under normal operations and enhanced tolerance during hypothetical accident scenarios, with implementation of batch reloads of one or more near-term ATF concept(s) in commercial reactor(s) in the 2023-2026 timeframe;
2. collaborate with industry to perform the R&D necessary to support extending the burnup of current commercial LWR fuels from 62 to 75 GWd/MTU by 2026;
3. lead research and development on innovative fuel and cladding technologies with applications to future advanced reactors, especially metallic fuels for fast-spectrum reactors, including reactors that utilize both once-through and recycle scenarios;
4. continue the development and demonstration of a multi-scale, science-based approach to fuel development and testing, and contribute to the establishment of

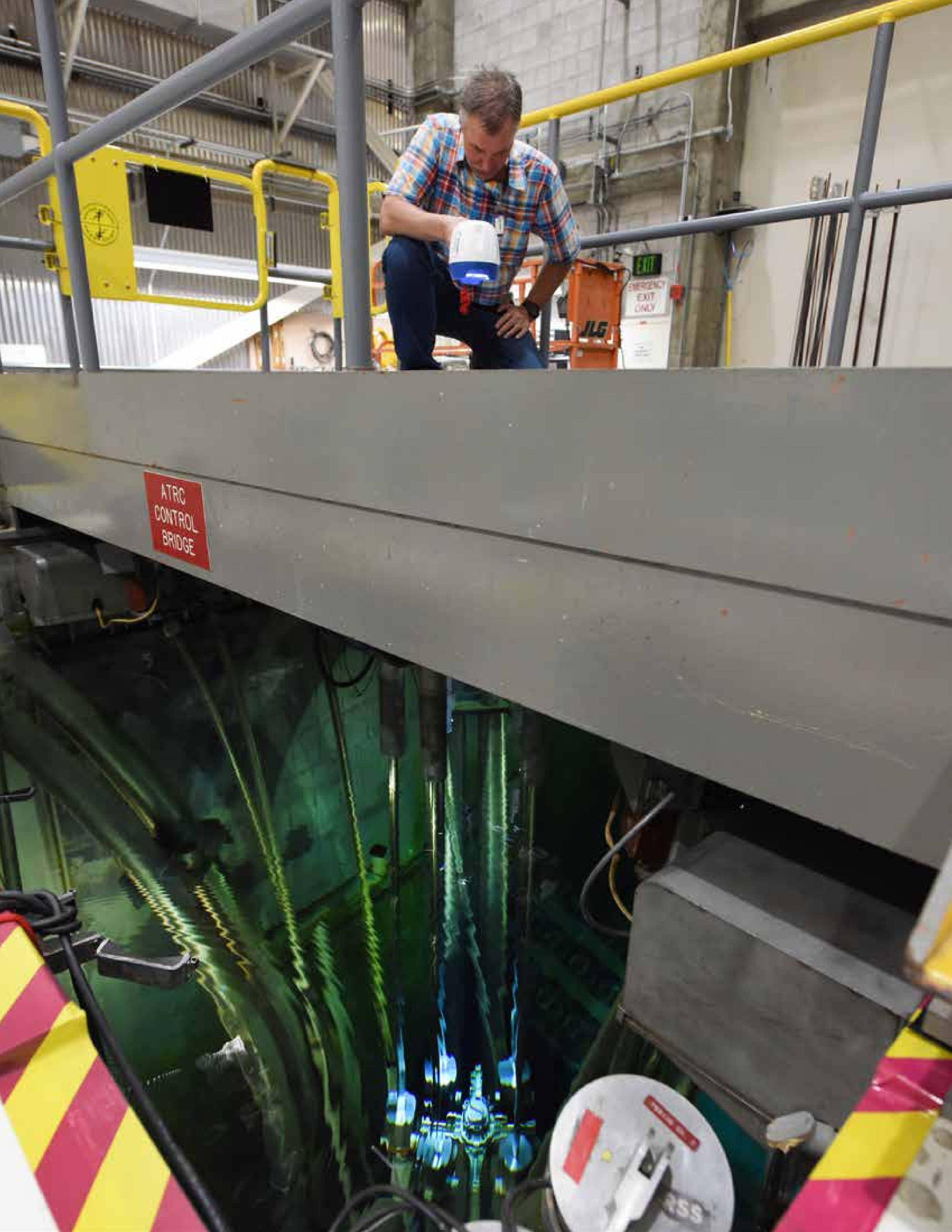
a state-of-the-art R&D infrastructure necessary to accelerate the development of new fuel concepts; and

5. collaborate with NEAMS on the development and validation of multi-scale, multi-physics, and increasingly predictive fuel performance models and codes.

Two developments of special note during FY20 were the following. First, with the Congressional appropriation bill passed in December 2019, funding for the R&D activities associated with advanced reactor fuel development was essentially eliminated. Some accomplishments had already been achieved during the first quarter of the fiscal year, and some funding was provided for an orderly close out major activities in progress. However, the portion of the Accomplishments Report dedicated to the Advanced Reactor Fuel Development Focus Area is considerably smaller this year for that reason. Second, the structure of the campaign was significantly reorganized in order to create better alignment with the new funding realities and shifting R&D priorities. There are now Qualification Leads in the areas of both Accident Tolerant Fuels and High Burnup Fuels to guide near-term activities, interact more directly with industry FOA teams, and coordinate with important stakeholders (e.g., EPRI, NRC). It is hoped that such Qualification Leads can be established for advanced fuels product lines in the future, as funding for such activities is secured. In addition, a number of Strategic Leads were established to take a longer view on fuels R&D needs of the future and identity those fuel technologies most appropriate for maturation to the point where they might become formal qualification programs. Those interested in learning more about the new structure will find it presented in the FY20 update to the AFC Program Execution Plan.

This report provides concise summaries of many of the significant AFC accomplishments made during FY20. Of particular note are the following key accomplishments and their significance:

- ATF-2 completed 244 EFPDs of irradiation in the ATR pressurized water loop with Westinghouse, Framatome, and General Electric test rods. Three new twelve-inch-long test rods were fabricated for Westinghouse and installed in a newly designed tier. The new tier increases the number of rods that can be irradiated.
- After ~184 effective full-power days of irradiation, ATF-2 was reconfigured to ATF-2B1 and reinstalled into the Advanced Test Reactor 2A Loop to resume irradiation. The ATF-2B1 configuration included new BWR-type ATF rodlets supplied by GE in Tiers 3 and 4. The PWR-type rodlets in Tiers 1, 2, and 5 are now operating at more prototypical pressure and temperature.
- Fabrication and assembly of the first MiniFuel capsules containing monolithic specimens of UO_2 and U_3Si_2 fuels was completed and began irradiation in HFIR Cycle 487, with target burnup levels of 8-10 and 28-40 GWD/MTU.
- The activity to install an I-Loop in ATR for prototypic ramp and run-to-failure testing of PWR and BWR fuel rods became a formal project, and fabrication began on a new Top Head Closure Plate to be installed during the Core Internals Changeout (CIC) in FY21. This new testing capability is needed to support qualification of both ATF and high burnup fuels for commercial reactors.
- Grain growth kinetic models were developed for undoped UO_2 in order to provide grain sizes similar to those in doped UO_2 fuels, which will be used in synthesizing materials to be used as reference standards for fission gas release studies.
- A biaxial flexure strength test was developed and qualified for use on UO_2 and other nuclear fuels to aid in understanding cracking and attempts to mitigate it, especially in high burnup fuels. Future work will employ this system to characterize doped UO_2.
- A revision of the Handbook on the Material Properties of FeCrAl Alloys was issued, with a significant focus on updating the experimental database to include the alloy C26M, a leading ATF alloy deployed in lead test rods in commercial reactors.



ATRC
CONTROL
BRIDGE
EXIT
EMERGENCY
EXIT
ONLY
JLG

- PIE results were published on UN kernels and UN-coated particles irradiated up to 10 GWd/MTU in only 68 EFPDs in HFIR using a MiniFuel capsule to accelerate burnup accumulation. Characterization of fission gas release and microstructural changes indicated the performance of these fuels was good under test conditions.
- A series of commissioning tests (ATF-RIA-1A,B,C,D) were performed in TREAT using the Minimal Activation Retrieval Capsule Holder-Static Environment Rodlet Transient Test Appartatus (MARCH-SERTTA) device to validate simulations of the design basis accident (RIA) and qualify new in-situ instruments.
- The first TREAT experiment on (4) ATF fuel rodlets, namely U3Si2 fuel pellets clad in both Zircaloy and SiC, was completed. The test was instrumented with both thermocouples and pyrometers, and measured temperatures compared well with pre-test predictions. Information obtained from the test will be used to begin development of fuel safety criteria for the new fuel in design-basis reactivity-initiated accident conditions.
- Remote assembly of a MARCH-SERTTA capsule in HFEF demonstrated a new capability to prepare previously irradiated fuel rod/rod segments for irradiation testing in either ATR or TREAT.
- Thermal cycling experiments were performed in the Severe Accident Test Station (SATS) on C26M and Zircaloy-2 tubes to gauge accident tolerance of these cladding materials under simulated cyclic dryout conditions. Results indicate that C26M possesses superior high temperature oxidation resistance, mechanical properties, and resistance to fatigue.
- A report was issued documenting an evaluation of reactor performance and safety impacts associated with the use of increased enrichment fuels in commercial power reactors. Results showed that increasing enrichment reduces the quantity of high-level waste disposed per unit energy generated, but increases the natural resource requirements normalized to a gigawatt-electricity-per-year basis. Additionally, the slightly higher discharge burnup results in somewhat different activity levels of the spent nuclear fuel and high-level waste radioactivity at 100 and 100,000 years after fuel discharge. Ultimately, no neutronic or reactor safety hindrances to employing light water reactor fuel with enrichments greater than 5% were identified.
- The first FAST (Fission Accelerated Steady-state Test) experiment, including metallic fuels with sodium-free annular geometry, minor alloy additives for lanthanide fission product control, and cladding liners as diffusion barriers against fuel-cladding chemical interaction, was fabricated and assembled and will be installed in ATR Cycle 169A in early FY21.
- PIE was initiated, and NDE completed, on a rodlet discharged from AFC-4C that had been irradiated to 8.7% burnup; the rodlet had U-10Zr fuel sodium-bonded to HT9 cladding that had an internal FCCI barrier of chromium applied by Korean Atomic Energy Research Institute (KAERI).
- A focused ion beam(FIB)/transmission electron microscopy (TEM) examination was performed on a sample from a legacy EBR-II transmutation fuel pin (X501). This TEM examination on irradiated metallic fuel is the first of its kind and sheds light on the microstructural changes in irradiated metallic fuel caused by the addition of minor actinides.
- The first examination of irradiated fuel (U-10Zr-1Pd from AFC-3A) was non-destructively examined with pulsed neutrons at the Los Alamos Neutron Science Center (LANSCE); analysis of the results and comparison with destructive examinations will attempt to validate this NDE method for characterizing lanthanide fission product migration and interaction with the cladding.
- The Advanced Reactor Experiments for Sodium Fast Reactor Fuels (ARES) collaboration was established with Japan to investigate transient fuel performance of advanced and high burnup fuel designs using legacy oxide and metallic fuel pins previously irradiated in EBR-II.

1.3 SHOWCASE CAPABILITIES

Irradiated Fuel Rod Refabrication

Principal Investigator: Jason Schulthess
Team Members/ Collaborator: Spencer Parker, Evans Chambers, Jordan Argyle, Kim Davies, Mark Cole, Colby Jensen, Cad Christensen and Gene Matranga

Figure 1. Out of cell demonstration of rodlet end preparations for welding, specifically, removal of the exterior oxide layer for welding.

Irradiated fuel rod refabrication is a crucial enabling capability that bridges between base reactor irradiation and subsequent experimentation and/or re-irradiation. It is key to performing meaningful research and development (R&D) on fuels with any level of burnup, especially at the Transient Reactor Test (TREAT) Facility, and opens the door to R&D for materials tested in commercial nuclear power plants. Several industry-led projects already exist that are dependent upon refabrication capability to generate data to support licensing for Accident Tolerant Fuels (ATF) and extension of fuel burnup limits. Refabrication allows access to fuel at any point in its lifetime, allowing opportunity to apply instrumentation and perform experiments to measure performance under a variety of specified conditions. Measurements on high burnup fuels are impractical, if not impossible, without refabrication capability.

Project Description:

A fuel refabrication system has been designed and constructed to be installed in the Hot Fuel Examination Facility (HFEF) hot cell to enable basic fuel rod refabrication. The basic system will include limited options to "reinstrument" irradiated fuel, which is currently planned as an advanced refabrication and reinstrumentation system to work in concert with the basic refabrication capability. The basic refabrication capability includes segmenting full length fuel rods, defueling the ends to make space for new end caps, attaching and welding in place new end caps, and pressurizing and seal welding the new rods, followed by leak check inspection, which are then available for follow on irradiation or out of pile testing. Installation of the basic fuel rod refabrication system is planned for fiscal year (FY) 2021.

The basic refabrication welding systems have been designed to provide maximum flexibility and reliability to perform the three necessary welds. Two circumferential welds to attach the end caps, and one seal weld to ensure hermeticity of the rodlet. All three welds are performed using commercially established micro-Tungsten Inert Gas (TIG) welding methodology. The circumferential welds are performed using a custom designed welding lathe, which holds the welding torch in a single position and rotates the rodlet, which provides the most consistent welding on these materials. The lathe system further incorporates an automated variable voltage controller, which supports consistent welds if the rods are found to be non-circular (e.g., elliptical). The seal welding system also utilizes micro-TIG methodology and encapsulates the rodlet in a pressurized chamber capable of pressurizing the rodlet from 500 psi to 2250 psi to allow for experiments to be conducted with variable rodlet internal pressures.

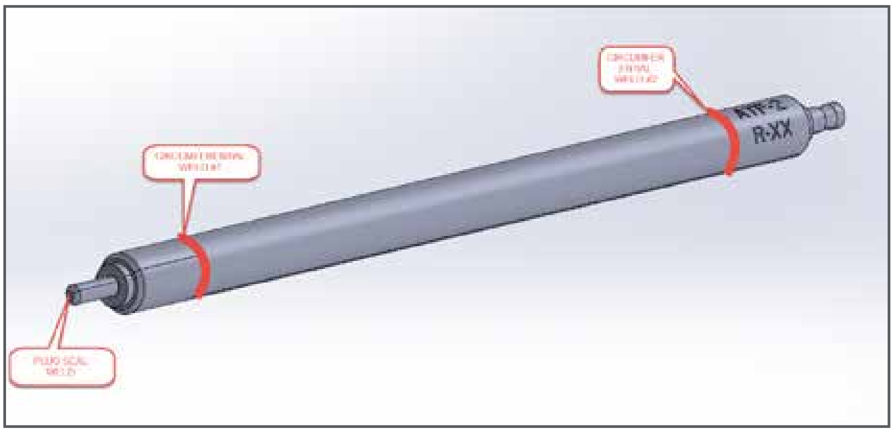


Figure 2. Three welds for basic refabrication capability are shown. Two circumferential welds and a single seal weld completed under pressure.

Accomplishments

During FY2020, out of hot cell methodologies were established for performing bulk mechanical defueling, fuel chemical dissolution, rodlet end preparations for welding, circumferential welding of end caps, pressurized seal welding, and post welding leak inspection. This included designing, fabricating and assembling equipment hardware, equipment control systems, establishing operational procedures, and performing out of cell functional and verification testing. In addition, basic facility infrastructure needs were identified and in some cases necessary design was completed such as for a new hot cell wall penetration feedthrough to support the welding system power and controls cables. In other cases, existing systems were identified and restored to functionality such as the HFEF-compatible helium leak-check system which is used to perform post seal welding inspection.

For mechanical defueling, the existing HFEF in-cell 3-axis end mill was selected for use. An out-of-cell demonstration was performed and identified the need to develop a debris collection system to meet facility requirements for collecting and accounting for removed material. A vacuum system was developed for this purpose and is currently being fabricated. Once fabrication of the vacuum system is completed, an in-cell mechanical defueling demonstration can take place.

Wet/chemical defueling utilizes an acid bath to dissolve the fuel and remove it from the cladding. This process uses 6M HNO3 heated to near

A new refabrication capability has been developed to support follow on irradiation and out of pile testing of irradiated fuel.

Figure 3. Circumferential welding system, currently undergoing out of cell function, verification, and demonstration testing.

boiling (~150 C). Once at temperature, the fuel sample is introduced to the system, and the acid etches the fuel out of the cladding. Once dissolution is completed, the acid bath will be neutralized with MgO to eliminate the hazardous nitrates and then solidified using a WIPP BoK approve sorbent. A demonstration of this procedure was performed at the Analytical Laboratory using a 5 mm length tubular segment of an irradiated fuel rod from the H.B. Robinson plant. a technical evaluation was initiated and is ongoing to evaluate chemical dissolution of larger segments within HFEF. This evaluation encompasses the impacts to both the facility and other programs to accommodate a larger acid bath dissolution system, and the necessary equipment and/or facility changes needed to accommodate the operation. A specific example is the evaluation of the vaporized fumes of nitric acid emanating from the heated acid bath, and the impact of those fumes on the facility and other programs, and if it is necessary to capture and neutralize those fumes.

Once fuel has been removed from fuel rod segments, final preparations for refabrication may begin. Refabrication includes welding on new endcaps to the fuel segment, which requires that the end be free of any oxide layers (to support welding arc initiation), that the ends be faced orthogonal to the axis of the segment, and that the end be free of any burrs or other features which may inhibit the installation of new endcaps. The approach selected was to use the available remotized 3-axis end-mill in the HFEF hot cell. However, the machining operations to remove the oxide layer require the installation of a 4th rotation axis. In this case the fuel segment would be placed into a rotating stage so that the

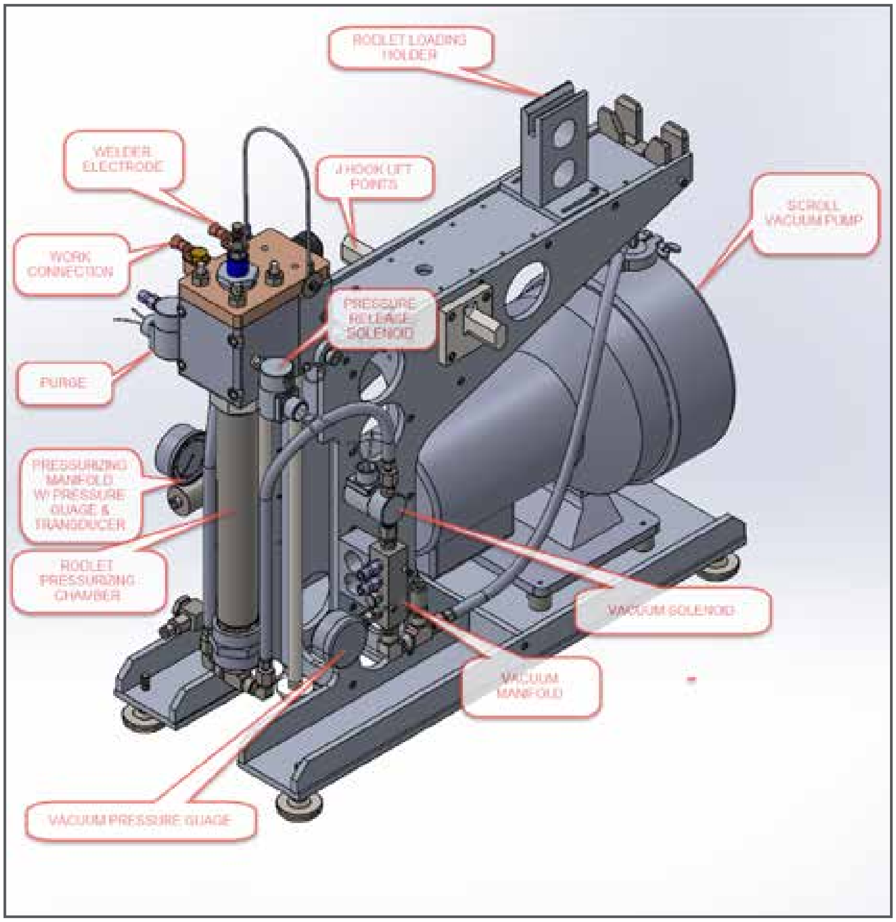


Figure 4. 3D model rendering of the design of the weld under pressure system, which is currently undergoing final assembly prior to out of cell function, verification, and demonstration testing.

fuel segment can be rotated beneath the end-mill cutting tool. A compatible rotating stage was procured and out of cell demonstration and trials have been performed to verify functionality and operation. Currently, this rotating stage is undergoing remote mockup testing to verify compatibility for remote operations. Once completed, the new rotation stage can be installed in the HFEF hot cell, and final demonstration and checks performed.

Two micro-TIG welding systems have been designed and constructed to perform the circumferential welds and seal welds necessary to attach new endcaps and set the desired rod internal pressure prior to follow on experiments. This includes completing the design for a new hot cell penetration feed through to supply the utilities and control system for the welding operations. The HFEF leak system was identified and restored to operation.

During the next fiscal year, the new equipment will be installed in the HFEF hot cell and demonstrated, which will showcase its function and availability by late FY2021. The first use of the crucial refabrication system is planned for fuel being shipped to Idaho National Laboratory (INL) from the Byron Nuclear Power Plant, which is a significant milestone.

HFIR Irradiation of Coated-Zr ATF Cladding

Principal Investigator: Kory Linton
Team Members/ Collaborator: Patrick Champlin, Padhraic Mulligan, Chris Petrie and Annabelle Le Coq

Coated cladding experiments in ORNL's High Flux Isotope Reactor (HFIR) are being leveraged to accelerate ATF qualification.

The understanding of radiation effects on coated zirconium accident tolerant fuel (ATF) cladding concepts is being accelerated using large-scale science tools like the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory (ORNL). ATF vendor teams leveraging various HFIR irradiation vehicles such as "rabbits" or full-length targets can quickly achieve light water reactor (LWR)-relevant temperatures and neutron fluences for a wide range of cladding sample lengths and geometries. After irradiation, the sample coating performance can be evaluated in hot cells and low-level radiation facilities to collect data on coating adhesion and mechanical properties, including fatigue, creep and simulated loss-of-coolant-accident (LOCA) burst behavior.

Project Description:

HFIR irradiations of cladding samples and creep tubes provide a unique opportunity to accelerate separate effects testing of coated zirconium ATF concepts by attaining end-of-life fluences in about 8 cycles, or approximately one year in-core. Rabbit capsules are 64 mm long aluminum cylinders used for irradiating a variety of samples in HFIR. For the coated cladding irradiations, the capsules are loaded in a target rod rabbit holder (TRRH) or peripheral target position (PTP) in the flux trap located at the radial center of the reactor core – exposing the samples to a nearly constant fast neutron flux of ~1×1015 n/cm2s during each ~24-day cycle. Full-length targets can also be inserted into the flux trap, allowing for irradiation of longer cladding samples (~100 mm). The three irradiation vehicle designs currently being utilized to explore ATF coatings are the "FlexClad" rabbit, full-length cladding target, and pressurized creep tube rabbit.

FlexClad

The "FlexClad" rabbit capsule design for standard pressurized water reactor (PWR) geometries accommodates three different specimen geometries in two configurations at a target temperature of 350°C. The rabbits hold either two 23.45 mm long standard PWR geometry cladding specimens (9.50 mm outer diameter) with molybdenum inserts [1],[2] or up to four 12 mm long standard cladding, axial tension, or plane strain specimens in a single rabbit with zirconium inserts (Figure 1). The axial tension geometry features an upper and a lower ring connected by two gauge regions, whereas the plane strain geometry is a standard cladding tube with two indentions at the top and the bottom. Both experiment configurations achieve the target temperatures by controlling the size of the specimen insert (i.e., gamma heating) and the specimen-to-housing insulating gas gap. Thermal analyses have been performed to determine temperatures at both beginning-of-life (BOL) and end-of-life (EOL) neutron fluences

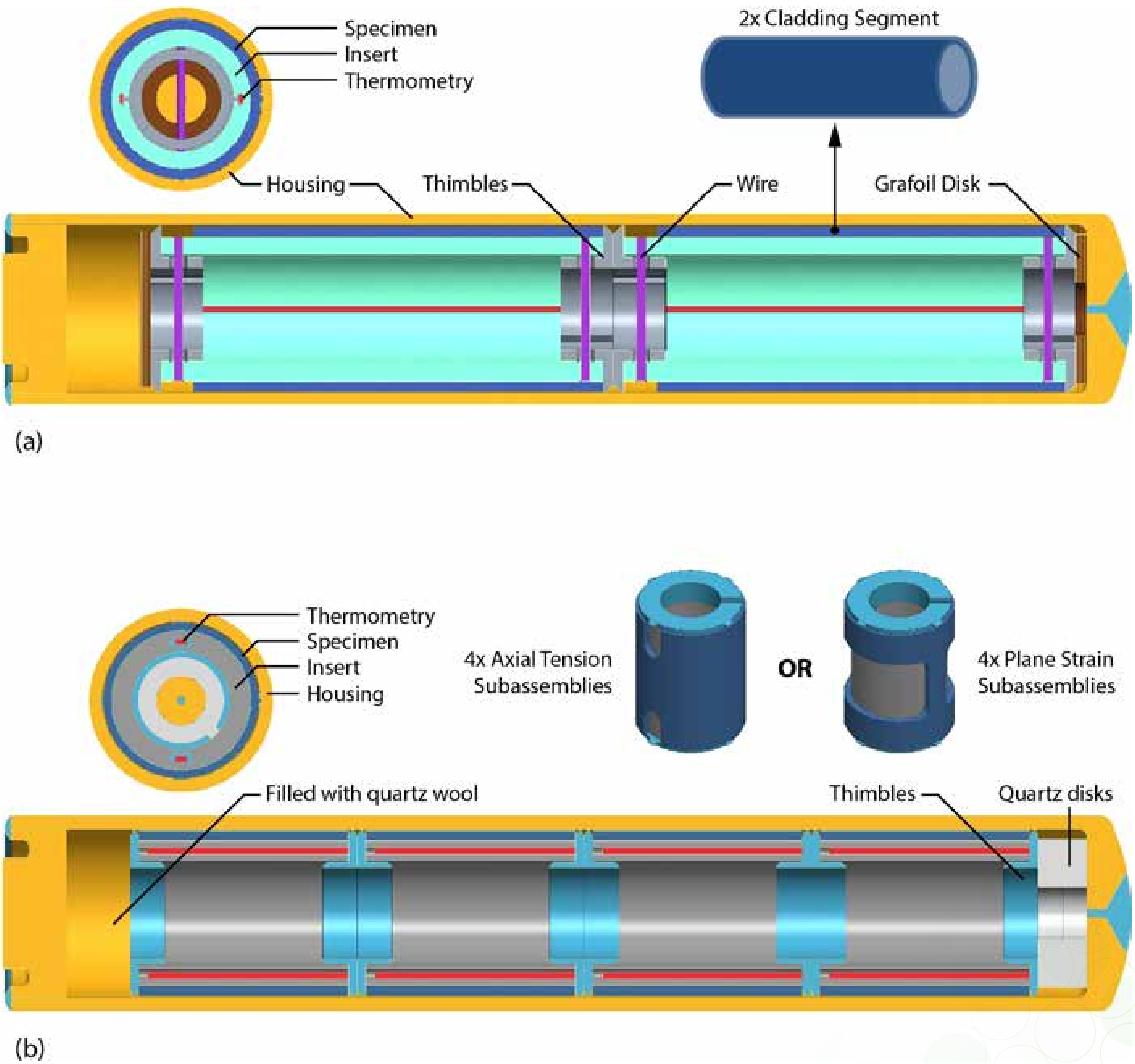


Figure 1. Design of the capsule containing (a) 2x - 23.45 mm long cladding specimens and (b) 4x – 12 mm long standard, axial tension or plane strain specimens [3].

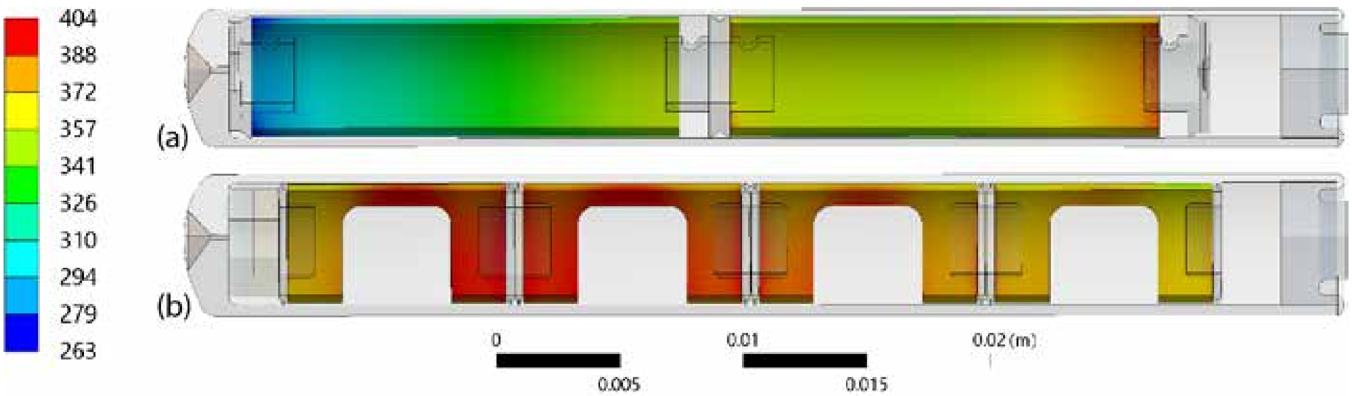


Figure 2. Temperature contours (°C) for the (a) FlexClad and (b) axial tension specimens at end of life (8 cycles) [3].

due to dimensional changes resulting from irradiation growth of the zirconium cladding. These analyses are used to target time averaged, volume averaged specimen temperatures in the 300-400°C design window (Figure 2). [3].

Full-length Cladding Target

Hot cell fatigue and burst testing requires cladding sample geometries much longer than what can be accommodated in rabbit irradiation vehicles, so a full-length cladding target has been designed to irradiate

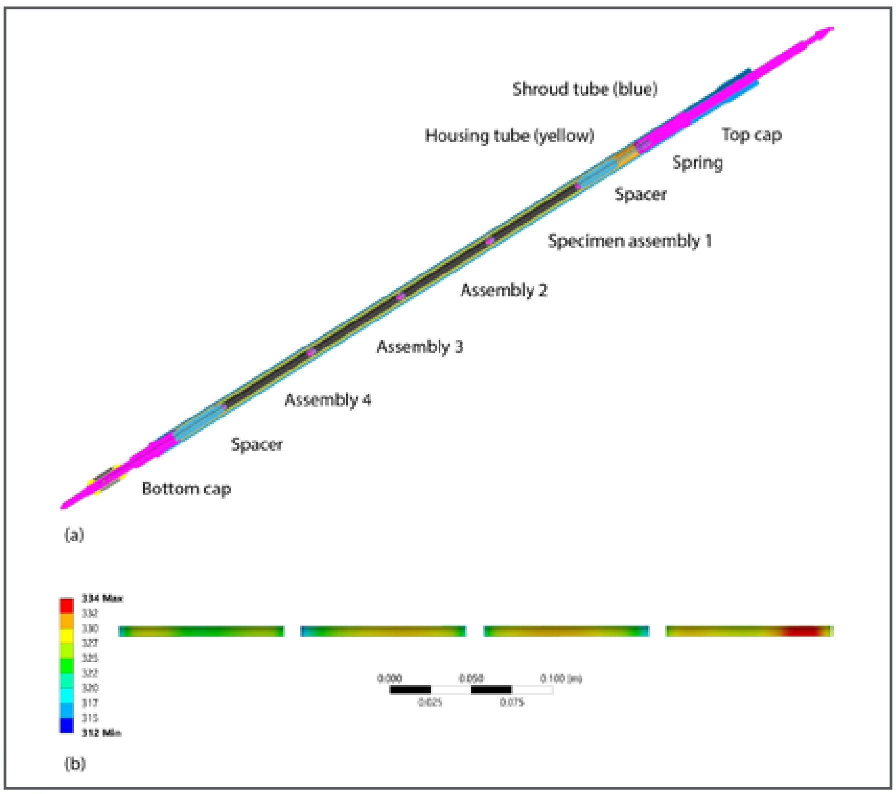


Figure 3. (a) Design of the 100 mm long cladding irradiation experiment [2]. (b) Temperature contours (°C) for the 100 mm long cladding tube specimens during HFIR irradiation [4].

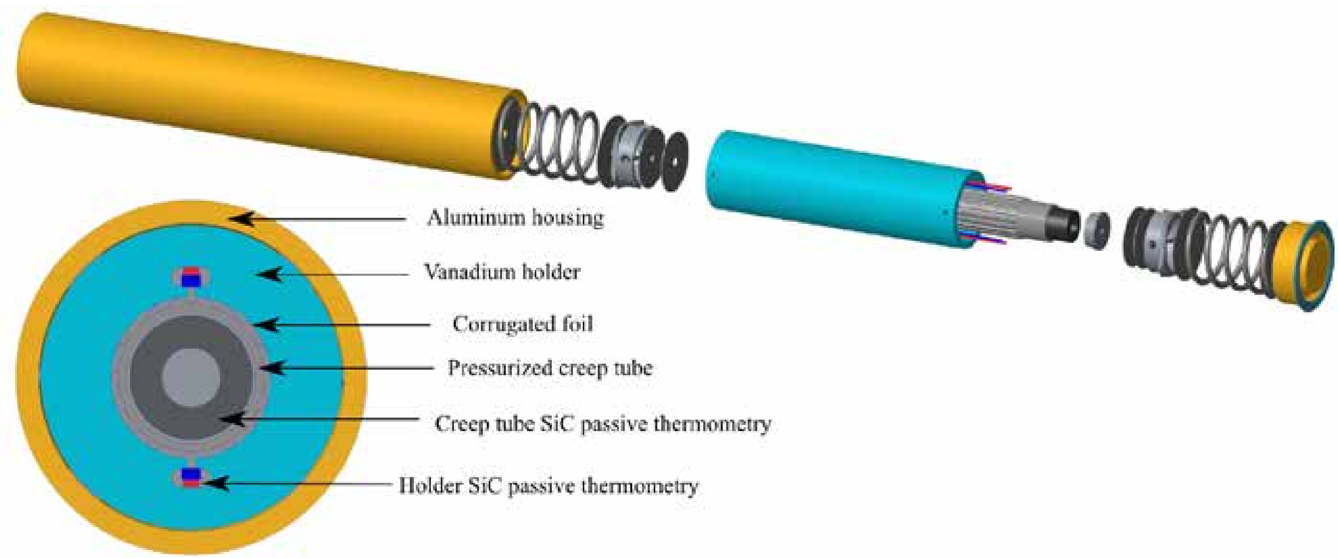


Figure 4. Design of the creep tube irradiation experiment [4].

four 100 mm long coated cladding specimens with a target temperature of 300 – 350°C (Figure 3). Spacers align the assemblies to be centered around the midplane of the reactor core and a tapered aluminum holder counteracts the reduction in nuclear heating rates further away from the axial midplane of the reactor core, allowing for more uniform temperatures throughout the length of the specimen. The single HFIR target with four specimens produces two dose conditions: one for the two specimens in the centerline region with minimal dose variation across the specimens, and one for the specimens loaded in the outermost target positions with larger dose variations. The full-length irradiation vehicle design combined with prior research demonstrating integral LOCA burst testing on 100 mm long specimens [2] allows for in-cell integral LOCA burst testing or fatigue testing of up to 4 specimens per target and 2 dose conditions.

Pressurized Creep Tube Targets

Rabbit capsules were designed (Figure 4) and fabricated (Figure 5) to irradiate pressurized creep tube specimens made from coated zircaloy. While irradiation creep of standard zirconium fuel cladding is well characterized, the effect of coatings on cladding creep has not been quantified. To address this data gap, thin-walled zirconium rodlets were coated with varying thickness and internally pressurized to generate a hoop stress in the rodlet. Depending on the internal pressure, this stress can be as high as 190 MPa. An embossed, compressible zirconium foil was placed around the pressurized rodlet to maintain a constant heat transfer path–and therefore constant ~300°C temperature–despite dimensional changes in the rodlets due to irradiation growth and creep. The diameter of each rodlet is precisely measured before irradiation using a customized 2-dimensional non-contact scanning

Figure 5. 2-D non-contact scanning profilometer (left) and pressurized creep tube capsule prior to assembly (right).

profilometer [3]. Measurements of irradiated rodlets will be repeated to quantify creep as a function of coating thickness, dose, and stress. Future work includes out-of-pile thermal creep measurements on similar coated specimens, and additional irradiation creep specimens irradiated to higher fluences.

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[6.] J. Nucl. Mater., Vol. 15, May 2018 (254-260).

ADVANCED LWR FUELS

2.1 ACCIDENT TOLERANT FUELS

Advanced LWR Fuel Advisory Committee

Committee Chair: Bill Gassmann, Exelon
Collaborators: Steven Hayes, Ed Mai, and Kate Richardson

The Advanced LWR Fuel Advisory Committee was established in 2012 to advise the Advanced Fuel Campaign (AFC's) National Technical Director on the direction, development, and execution of the campaign's activities related to accident tolerant fuels for commercial light water reactors (LWR). Last year the committee charter was revised, and this year the committee was asked to provide an industry perspective on LWR fuels issues that are broader in scope than just accident tolerant fuels. The Industry Advisory Committee (IAC) is comprised of recognized leaders from diverse sectors of the commercial light water reactor industry. They represent the major suppliers of nuclear steam supply systems, owners/operators of U.S. nuclear power plants, fuel vendors, Electric Power Research Institute (EPRI), and Nuclear Energy Institute (NEI). Members are invited to participate on the committee based on their technical knowledge of nuclear plant and fuel performance issues as well as their decision-making authority in their respective institutions. During the past year the committee provided important industry input relative to utility and fuel vendor perspectives on the potential benefits of extending the burnup of current fuels; continued efforts in testing and evaluation of new accident tolerant fuels, especially relative to the lead test assemblies operating in numerous commercial plants that include Accident Tolerant Fuels (ATF) rods; testing infrastructure needs and gaps created by the loss of the Halden Reactor in Norway; and development of testing plans for the QUENCH facility in Germany.

The IAC meets monthly via teleconference and is currently chaired by William Gassmann of Exelon Corporation. Additional members represent Westinghouse Electric Company, Global Nuclear Fuels, AREVA, Dominion, Duke Energy, Southern Nuclear, EPRI, and NEI.

AREVA

Exelon

NEI NUCLEAR ENERGY INSTITUTE

Dominion Energy

EPRI ELECTRIC POWER RESEARCH INSTITUTE

Southern Company

DUKE ENERGY

GNF Global Nuclear Fuel
A Joint Venture of GE, Toshiba, & Hitachi

Westinghouse

ATF Industry Teams - Framatome

Principal Investigator: Kiran Nimishakavi
Team Members/ Collaborators: Idaho National Laboratory (INL), Oak Ridge National Laboratory (ORNL), Southern Nuclear Operating Company, Kernkraftwerk Gösgen-Däniken. Entergy Nuclear, Exelon Nuclear.

Figure 1. Green chromium variant ATF pellets before sintering.

Framatome's project objective is to maximize plant safety by providing evolutionary fuel technologies in the near term that adds margin to a number of fuel performance areas while also developing a longer-term revolutionary solution.

Framatome is continuing significant research and development efforts to develop accident tolerant fuel technologies. It relies on a two-phased approach enabling the balancing of benefits with the speed to market. The first phase consists of near term evolutionary solutions which are focused on improving safety and fuel cycle economics while being compliant with insertion of batches by 2025. The second phase consists of longer-term solutions aimed to offer significant improvements during beyond design basis accidents.

In addition, Framatome's Advanced Fuel Management (AFM) initiative is focused on extending the length of the current fuel cycles in PWRs from 18 to 24 months. The objective is to take advantage of fuel enrichments above 5% to increase energy production and reduce outage costs by reducing the number of refueling outages while simultaneously defining appropriate licensing limits for burnup.

Project Description:

The ultimate goal of Department of Energy (DOE's) Enhanced Accident Tolerant Fuel (EATF) program is to develop an economical and more robust nuclear fuel design that will reduce or mitigate the consequences of reactor accidents while maintaining or improving existing performance and reliability levels in daily operations. After extensive testing, evaluation and downselection during the program's first phase, Framatome's Phase II technical approach addresses three focus areas: (i) Chromium (Cr)-coated cladding, (ii) Chromia-doped UO_2 fuel pellets, and (iii) Silicon carbide (SiC) composite materials.

A dense Cr-coating on a zirconium-based cladding substrate has the potential for improved high temperature steam oxidation resistance and high temperature creep performance, as well as improved wear behavior. Over the course of the EATF program, extensive processing and testing activities are being carried out in support of delivering EATF Lead Test Assemblies (LTAs), with two sets of LTAs successfully delivered in 2019, and support of batch implementation by the mid-2020s.

Chromia-doped UO_2 pellets can improve pellet wash-out behavior after cladding breach and reduce fission gas release. To date, the performance of this fuel has been extensively studied in out-of-pile and in-pile test programs and modifications are being implemented to accommodate chromia-doped fuel in Framatome's fuel performance code.

For revolutionary (over-the-horizon) performance improvements, Framatome is developing a composite cladding comprised of silicon carbide fiber in a silicon carbide matrix (SiC_f/SiC) as well as studying applications for other critical components such as BWR channel boxes and structural tubes. The objective is to develop a fuel system which does not suffer from the same rapid oxidation kinetics of zirconium-based cladding while having attractive operating features such as reduced neutron absorption cross-section and higher mechanical strength at accident temperatures.

Accomplishments:

A major emphasis of Framatome's ATF pellet development has been improving pellet thermal properties, especially thermal conductivity. Variants of the chromia-doped UO_2 fuel pellet have been in development in the Framatome GmbH Fuel Laboratory in Erlangen, Germany (see Figure 1). Testing shows a significant increased thermal conductivity compared to UO_2.

In-pile testing of new variants of chromia-doped UO_2 fuel is planned in the Advanced Testing Reactor (ATR) at Idaho National Laboratory (INL). A total of nine rodlets, which includes rodlets with standard UO_2 and two variants of chromia-doped UO_2 will be irradiated to study the evolution of thermal conductivity and microstructure under irradiation. Test rodlets were received by INL in the third quarter government fiscal year (GFY) 2020. These fuel rods will be placed in an outer capsule that provides isolation from the reactor water coolant. Capsules are scheduled to be inserted for irradiation testing in ATR cycle 169A.

Coating adhesion plays a critical role when considering the functional behavior of Cr-coated cladding. The adherence of the coating after irradiation was confirmed through Expansion Due to Compression (EDC) tests performed by the CEA at 350°C. This type of test is representative of the stresses imposed by the pellet on the cladding but with much higher deformations. Figure 2 shows the irradiated sample deformed up to 21% with no observed delamination of the Cr-coating. The apparent interfacial stability and absence of delamination clearly demonstrates that the coating retains excellent adhesion.

Figure 2. Irradiated Cr-coated tube after 21% deformation at 350°.

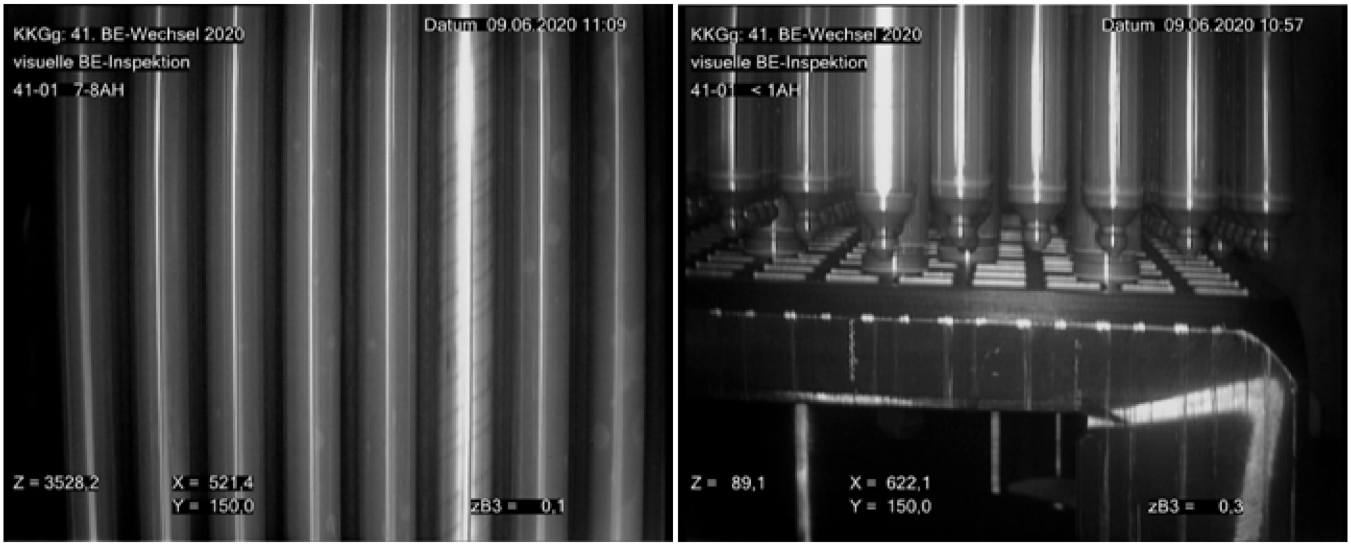

Figure 3. Visual inspection of Cr-coated LTRs in Gösgen (GOCHROM program).

Onsite visual inspections were performed on Cr-coated Lead Test Rods (LTRs) after a year of irradiation in the Gösgen reactor. The Cr-coating showed excellent adhesion with the underlying substrate and no signs of delamination were observed. As shown in Figure 3, a metallic bright appearance was clearly observed for Cr-coated cladding suggesting a significant reduction in corrosion kinetics relative to uncoated cladding.

In GFY 2020, Framatome completed fabrication of Cr-coated cladding tubes to support one full EATF LTA insertion in the Calvert Cliffs Nuclear Plant #2 in early 2021. This test program will allow for the evaluation of any interaction between a full bundle of EATF fuel rods and the rest of the fuel assembly structure and is a critical part of the overall batch licensing strategy.

Visual inspections were performed on four LTAs after first cycle irradiation in Southern company's Vogtle Unit-2. These assemblies contain four Cr-coated rods and all rods contain chromia-doped pellets. Cr-coated rods showed lustrous-gold appearance indicating a significant reduction in corrosion kinetics compared to uncoated rods.

Framatome is also developing an EATF coated cladding design for Boiling Water Reactors (BWRs) with a goal to introduce LTRs in a commercial reactor in 2021. Engineering work associated with LTR justification is currently ongoing.

SiC_f/SiC concept development is currently ongoing with a near-term goal of irradiation in Massachusetts Institute of Technology test reactor in early 2021. An environmental barrier coating has been identified to address the recession of the SiC_f/SiC concept (see Figure 4). A coating is applied on the external surface of the cladding and it is currently being tested in an autoclave under representative PWR conditions. Results after 70 days of

Figure 4. SiC_f/SiC composite tube with a metallic coating for LWR applications.

exposure show no degradation of the coating. Tests are still ongoing and more results are expected in the near-term. Framatome's multilayer SiC_f/SiC concept has also been designed to avoid any loss of hermeticity. The cladding design is being tested under different types of mechanical loads to evaluate its performance under normal and accident conditions.

Framatome signed a contract with General Atomics (GA) to evaluate the feasibility of SiC_f/SiC composite for BWR application. This work will involve manufacturing and testing activities that will lead to a better understanding of the challenges and opportunities for application in BWRs.

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Accident Tolerant Fuels (ATF) Phase II – General Electric Development of LWR Fuels with Enhanced Accident Tolerance

Principal Investigator: Raul B. Rebak, GE Research, Schenectady, NY
Team Members/ Collaborators: Russ Fawcett, Global Nuclear Fuels; Evan Dolley, GE Research; Andy Nelson, ORNL; Ed Mai, INL; Kenneth McClellan, LANL

GE is on track on obtaining actual commercial reactor plant irradiation data on two ATF concepts including ARMOR coated zirconium alloy and IronClad monolithic clad fueled rods.

As fuel vendors, General Electric (GE) including GE Research, GE Hitachi Nuclear Energy, and Global Nuclear Fuel (GNF) and their partners the reactor owners Southern Nuclear and Exelon Generation plus Oak Ridge National Laboratory (ORNL), Idaho National Laboratory (INL), and Los Alamos National Laboratory (LANL) are working in the development of accident tolerant fuels (ATF) for the current fleet of light water reactors (LWR). Activities include basic research and testing to characterize and evaluate advanced concepts that were never used before in reactor environments, fuel rod fabrication, as well as direct assessment of fuel rods behavior by installation is operating civilian nuclear power plants. The current GE contract with the DOE Office of Nuclear Energy DE-NE0008823 extends to February 2021.

Project Description:

The objective of the GE-led project is to develop a family of fuels that will make the current and future fleet of light waterpower reactors safer to operate. The newer family of ATF fuels will also add benefits such as; (a) Fuel cycle economics (i.e. increased burnup), (b) Increased fuel reliability, and (c) Plant operational flexibility (for example using power reactors for peak demands of electricity). GE is working in fuel concepts that are for near term implementation and for longer term development (advanced concepts). The fuel developments include cladding components, fuel components, and channels for boiling water reactors (BWR) applications. For cladding, GE is developing the ARMOR coating for Zircaloy-2 tubing which will provide outstanding resistance to fretting under normal operation conditions and increased resistance to oxidation in Design Basis Accident (DBA) and Beyond Design Basis Accident (BDBA) conditions. GE is also developing the IronClad cladding concept which involves the use of a monolithic FeCrAl alloy for housing the urania fuel. Since the current zirconium alloy used for channel materials needs to be replaced as well, GE is evaluating to utilize nuclear grade silicon carbide composite materials to fabricate the channels. Since the channel does not require hermeticity but requires stability in hot water and resistance to steam, the development of silicon carbide composites for channels is a logical first step before this material can be implemented for fuel cladding. On the fuel side GE is exploring the

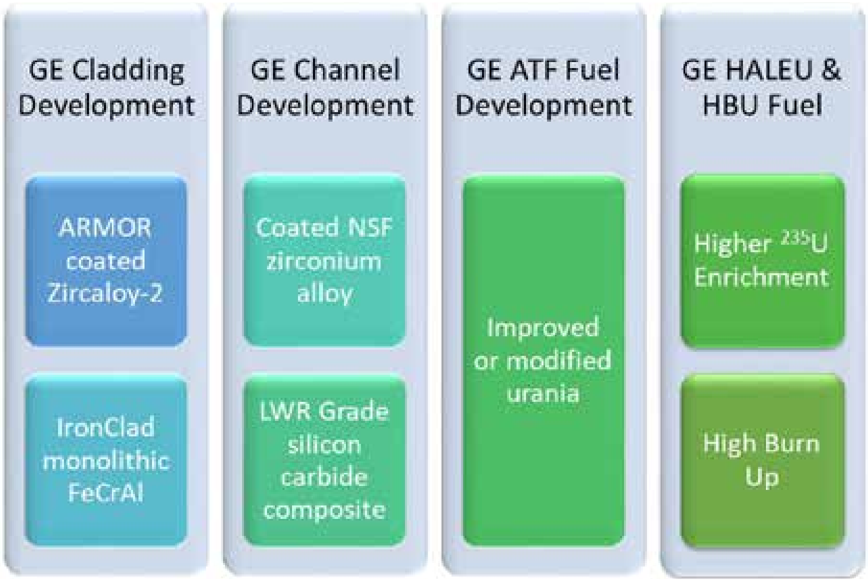

Figure 1. GE ATF 2020 Activities.

modification of the current urania fuel to make it more resistant to fragmentation in the case of an accident or for extended burn up conditions. GE is also concurrently working on higher 235U fuel enrichment and higher fuel burn-up concepts.

Accomplishments:

The two largest accomplishments for the FY2020 were; (1) The insertion of GE ATF fueled concepts into Exelon Generation's Clinton Unit 1 Cycle 20 Nuclear Power Station in October 2019 (Figure 1.), and (2) The Pool-side inspection of first ATF concepts inserted into a commercial power station at Hatch #1 in February 2020 (Figure 2.).

(1) The GE ATF concepts inserted into Clinton NPP included segmented full length rods which comprised fueled and non-fueled segments or ARMOR coated Zircaloy-2 and IronClad C26M as well as non-fueled segments of Advanced Powder Metallurgy Tubing(APMT) and Nippon Nuclear Fuel Development (NFD) Oxide Dispersion Strengthened (ODS) cladding.

(2) Southern Nuclear announced that the world's first installed ATF lead test rods have successfully completed a 24-month fuel cycle at the Edwin I. Hatch Nuclear Plant marking a significant milestone in the continued development of advanced fuel. Resulting data from this achievement will inform the advancement of a groundbreaking technology designed to increase plant performance and build upon an already robust margin of plant safety. During a planned maintenance and refueling outage, operators transferred a sampling of the GE GNF lead test rods from Plant Hatch Unit 1 to the spent fuel pool and have completed an initial inspection of

Figure 2. Clinton Unit#1ATF-LTA Installation October 2019.

the fuel in comparison to standard zirconium rods. Southern Nuclear's John Williams said that initial inspections have confirmed that the fuel performed as expected and that they anticipate leveraging this success and data with their fuel vendors into the continued development of this innovative technology. John Williams also said, "We will continue to pursue solutions like advanced fuel that enhance the performance and reliability of our operating plants and ensure the safety and health of our customers and our employees." ORNL will conduct further evaluations of the lead test rods' material and coating properties. The data obtained from this analysis will be used by Southern Nuclear and fuel vendor GNF to guide future development of ATF technologies and provide information to the Nuclear Regulatory Commission (NRC) licensing review process. The industry is pursuing the licensing and full commercial deployment of ATF by the mid-2020s.

The technical goals for the current FY2020 period were, (a) to advance on rod manufacturing procedures both for ARMOR coated and monolithic IronClad cladding, and (b) to make progress on obtaining irradiation data for the ATF concepts. Both goals were highly successful since it was demonstrated that standard industrial equipment could be used for fabrication of advanced concepts rods, and GE is continuing to explore the irradiation resistance of the ATF concepts at three sites simultaneously; (1) The ATF-2 program at the INL Advanced Test Reactor (ATR), (2) the poolside inspection at the Hatch Unit 1 plant in Georgia in February 2020 (which will continue with postirradiation examination (PIE) at ORNL), and (3) the newest commercial installation into Clinton Unit 1 plant in Illinois in October 2019.

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Accident Tolerant Fuel (ATF) and High Burnup - Higher Enrichment (HBHE) Fuel Industry Teams – Principal Investigator: Westinghouse Electric Company LLC

Principal Investigator: E. J. Lahoda

Team Members/ Collaborators: Westinghouse Electric Company LLC, General Atomics (GA), Massachusetts Institute of Technology (MIT), Idaho National Laboratory (INL), Los Alamos National Laboratory (LANL), Exelon Nuclear, University of Wisconsin (UW), National Nuclear Laboratory (United Kingdom) (NNL), Army Research Laboratory (ARL)/VRC/MOOG, University of Virginia, University of South Carolina, Oak Ridge National Laboratory (ORNL), Fauske & Associates, Rensselaer Polytechnic Institute (RPI), University of Texas at San Antonio, Air Liquide (AL)

Westinghouse EnCore® Fuel is "game-changing" for the nuclear industry, significantly increasing safety margins in severe accident scenarios, increasing flexibility for fuel management and enabling higher burnups for longer fuel cycles which can lower operating costs.

Westinghouse is working to commercialize unique accident tolerant EnCore®* fuel (ATF) designs with the capability of using higher U235 enriched ADOPT™* fuel or $U^{15}N$ fuel to achieve burnups of around 75 MWd/kgU; SiGA™* silicon carbide (SiC) cladding with higher U235 enriched fuel and Sn bonding or $U^{15}N$ fuel uranium nitride (UN) fuel.

Project Description:

Lead test rods (LTRs) of Cr coated cladding were loaded into the Byron Unit 2 reactor on April 23, 2019 and the Doel 4 reactor in July, 2020. Cr coated Zr claddings with ADOPT and UO_2 pellets were also loaded in the Advanced Test Reactor (ATR) in mid-2020. SiC clad tubes with Mo pellets will be loaded into the Massachusetts Institute of Technology Reactor (MITR) in early 2021. Cr coated Zr cladding and SiC cladding will be loaded into the BR-2 as part of the Il Trovatore program at the end of 2020.

The immediate tasks are aimed at design and licensing with the required experimental backup to obtain U.S. Nuclear Regulatory Commission (NRC) approval for insertion of lead test assemblies (LTAs) in 2021 or 2022 of Cr coated Zr with ADOPT pellets and with >5% enriched U235 pellets to increase achievable burnups to 75 MWd/kgU, allowing economic 24-month cycles in pressurized water reactors. Additional tasks include:

- Develop oxidation resistant UN and production technologies;
- Develop low cost methods for manufacturing SiC and Cr coated rods;
- High temperature out-of-reactor testing of Cr coated cladding to validate the MAAP and MELCOR codes to determine potential operational savings and to support licensing changes;
- Continue NRC licensing interactions aimed at implementation of regions in the mid-2020s.

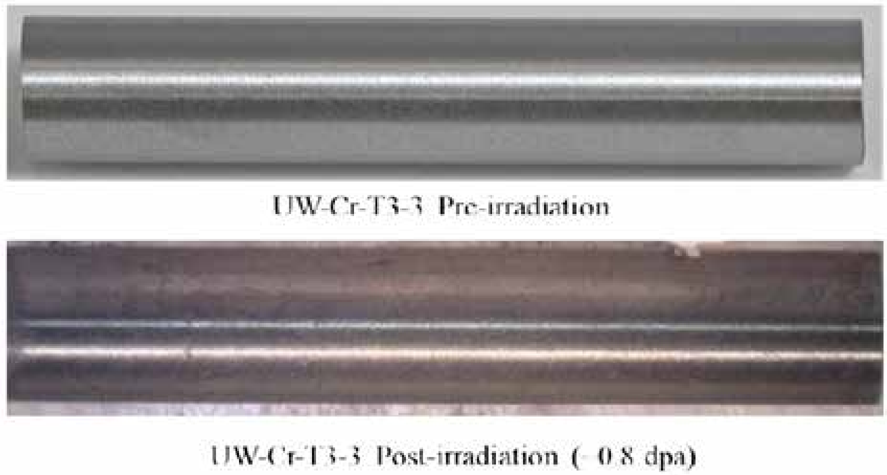


Figure 1. MITR irradiation of chrome coated zirconium cladding showed excellent performance.

Accomplishments:

A draft Topical Report for ADOPT fuel was sent to the NRC.

Coatings using physical vapor deposition (PVD) are being evaluated for mechanical properties and compared to cold spray (CS) applied coatings. Corrosion testing both in and out of reactor is being pursued. A cost study has concluded that PVD and CS costs are comparable. PVD offers an advantage over CS in that the Zr-Cr interface thickness variability is smaller and the neutronic penalty is therefore somewhat smaller.

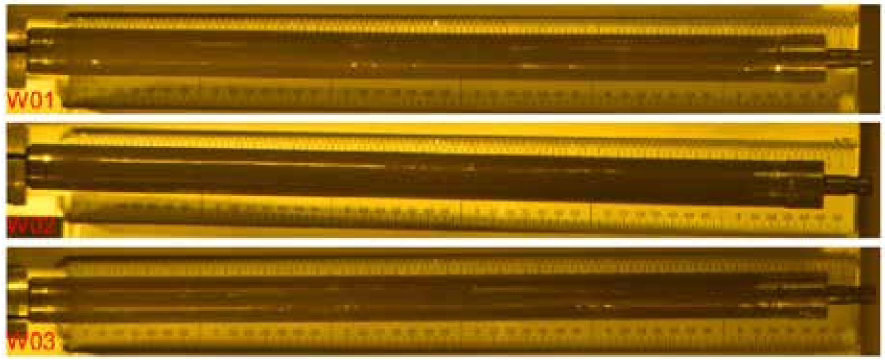
Efforts were initiated to determine the safety significant consequences of postulated fuel fragmentation, relocation, and dispersal (FFRD) in loss-of-coolant accident (LOCA) scenarios for the high burnup-higher enrichment (HBHE) program. Initial representative 3- and 4-loop high burnup and enrichment core designs were developed to facilitate investigations of fuel rod and safety analysis limits. Initial steps were taken to define the fuel rod characteristics which will facilitate high burnup / high enrichment core designs, including accident tolerant features.

General Atomics (GA) is continuing development of their SiGA clad fuel using a Sn liquid metal bonding technique. A detailed analysis is underway at GA to determine the root cause of the leak test discrepancy indicating that there is a feature unique to the third rodlet that has not yet been determined.

Tests are continuing at the MITR at pressurized water reactor (PWR) coolant conditions on the Cr coated Zr and SiC cladding options as well as the in-rod sensor that is being developed by Westinghouse to support the ATF testing and licensing process. The results indicate minimal corrosion of the Cr coated Zr samples and minimal to moderate corrosion of the SiC, depending on the manufacturing conditions of the SiC and the manufacturer.

LTRs containing EnCore cold sprayed Cr coated rods with UO_2 were delivered to Tractebel in late April 2020 for loading in Doel 4 during the summer 2020 outage.

Figure 2. Images of Cr cold spray coated cladding from rodlets W01, W02 and W03 after ATR irradiation.



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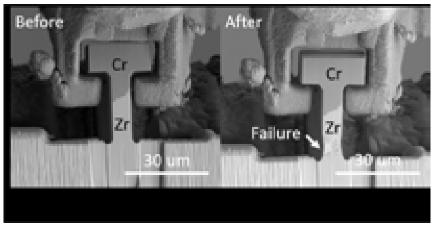


Figure 3. Micromechanical tests performed by INL showed that the Westinghouse Cr coating adhesion was stronger than the base Zr material.

University-Led Investigations of Thermal-Hydraulic Behavior of Accident Tolerant Fuel Materials

Technical Point of Contacts: Colby Jensen and Daniel Wachs
Team Members/ Collaborators: See individual project reports

Four university-led projects including 7 universities, 1 national laboratory, and 5 industry collaborators provide first investigations of two-phase heat transfer and critical heat flux for ATF materials.

In 2017, the Nuclear Energy University Program (NEUP) awarded 4 proposals to investigate two-phase heat transfer and Critical Heat Flux (CHF) for Accident Tolerant Fuel (ATF) materials. The projects worked closely to ensure complementary work in a exemplary manner. All four NEUP projects will finish at the end of this year, after providing first-of-a-kind insights and important data that will support ATF needs moving forward. A detailed summary report for the collection of works was published last year INL/EXT-19-56455 [1]. Figure 1 presents an overview of the project participants. The following provides a summary of the primary accomplishments of each project.

Project Description:

Project 17-12549

Title: Pool Boiling CHF Studies of ATF Cladding Materials with High Temperature Water Corrosion

Investigators: M. Corradini, H. Yeom, E. Gutierrez, K. Mondry, K. Sridharan, H. Jo (now Pohang University of Science and Technology, University of Wisconsin-Madison), P. Xu (now Idaho National Laboratory) and W. Byers (Westinghouse Electric Company).

Selected Highlights: Pool boiling experiments have been performed at atmospheric pressure on accident tolerant fuel cladding materials (Zirlo®, Cr coated Zirlo®, FeCrAl coated Zirlo®, and Chemical Vapor Deposition (CVD)Silicon Carbide (SiC)) to evaluate Critical Heat Flux. Selected results are presented in Figure 2. Corrosion resistant coatings of Cr and FeCrAl alloy were deposited on flat Zirlo® samples using cold spray technology. The as-prepared samples after surface polishing were subjected to autoclave tests at 360 °C water and 18.6 MPa for 360 hours to simulate prototypic corrosion of the ATF clad materials in an a light water reactor (LWR). Surface characteristics potentially influencing CHF such as surface morphology, roughness, static contact angle, and surface chemistry were characterized using a suite of characterization methods including scanning electron microscopy (SEM), 3D optical profilometry, optical contact angle measurements, and x-ray photoelectron spectroscopy. Thermo-physical properties of the samples such as density, thermal conductivity, and heat capacity were also measured. Polished FeCrAl coatings showed

Figure 1. Overview of four NEUP project participants studying thermal-hydraulic behavior of ATF materials.

CHF values comparable to bare Zirlo® samples, while slightly lower CHF values were observed for Cr coated samples and CVD SiC. As-deposited Cr coatings showed 67% higher CHF than the as-polished Cr coatings due to its higher surface roughness. CHF enhancement was generally observed for autoclave treated samples except for Cr coatings since minimal corrosion was observed post-test. Similar CHF values of bare Zirlo®, FeCrAl coating, and CVD SiC samples were observed after autoclave testing, but there was negligible change in the CHF for Cr coatings. XPS studies indicated that the formation or deposition of a few monolayers of hydrophobic carbonaceous species on surface can affect CHF values. The trends in CHF data are being analyzed in terms of the evolution of the surface characteristics of the ATF materials.

Project 17-12647

Title: Determination of Critical Heat Flux and Leidenfrost Temperature on Candidate Accident Tolerant Fuel Materials

Investigators: M. Bucci, B. Phillips, G. Su (Massachusetts Institute of Technology), M. Anderson, D. Lee (University of Wisconsin Madison), Z. Karoutas and Q. Wan (Westinghouse Electric Company)

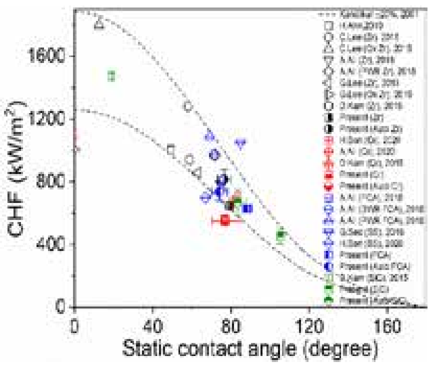


Figure 2. Selected results from UW-led project: CHF data for ATF candidate materials in pool boiling conditions as a function of static contact angle reported in literature and compare to present study. All tests were performed with flat samples in the horizontal direction at atmospheric pressure. Predicted CHF plots based on the Kandlikar CHF model with 20% uncertainty is shown.

Selected Highlights: The temperature effects on the wettability of Zircaloy, Chromium and FeCrAl using a first-of-a-kind apparatus developed to enable contact angle measurements up to pressurized water reactors (PWR) conditions, in a steam-saturated atmosphere. The results revealed that the intrinsic wettability of Chromium is slightly higher than Zircaloy or FeCrAl. However, at PWR conditions all materials are super-hydrophilic, i.e., their contact angle approaches zero.

Flow boiling tests at ambient pressure were performed using non-intrusive optical fiber and high-speed video diagnostics. Figure 3 presents examples of test results. The CHF on uncoated Zircaloy and Chromium coated Zircaloy is approximately the same (within a +/-10% uncertainty). Both PVD coated and spray coated Chromium were tested no major differences were observed. However, during the boiling crisis, the Chromium coated surfaces are more oxidation resistant than bare Zircaloy. This property is beneficial, as it seems to reduce the risk of melting.

The Westinghouse Advanced Loop Test (WALT) loop was used to conduct tests in PWR conditions. The tests have revealed that there is no significant difference in CHF between Chromium coated and bare Zirlo, even in the presence of CRUD. This suggest that the Chromium coating does not affect thermal margins. However, while these tests have been performed using a direct heating technique, future experiments will be used using an indirect heating, to verify that the observed phenomena are not driven by the heating solution.

Currently, plans are being finalizing for flow boiling tests in high pressure conditions, including preliminary transient scenarios, such as loss of flow and reactivity-initiated accident conditions (e.g., an exponential power escalation).

Project 17-12688

Title:

An Experimental and Analytical Investigation into Critical Heat Flux Implications for Accident Tolerant Fuel Concepts

Investigators: A. Prinja, M. Chen, Blandford (now Kairos), Y. Lee (now Seoul National University, University of New Mexico), N. Brown (University of Tennessee-Knoxville), W. Marcum (Oregon State University),C. Jensen (Idaho National Laboratory), John Strumpell (Framatome), Raul Rebak (General Electric)

Selected Highlights: Flow boiling critical heat flux (CHF) experiments indicate that the thermomechanical properties of materials affect CHF values for various claddings. The dependence of flow boiling CHF on cladding material is gradually reduced by increasing mass flux and/or inlet subcooling. Figure 4 presents a summary of mass flux and material effects on CHF behavior. Beyond the mass flux of 2000 kg/m2s and the inlet subcooling of 30 , the CHF differences contributed by cladding materials may be negligible. This implies that the current CHF database for stainless steels and other commercial alloys could be applicable to accident tolerant fuel (ATF) materials under certain conditions. The mechanism behind the CHF difference could stem from two competing mechanisms: 1) the material-dominated

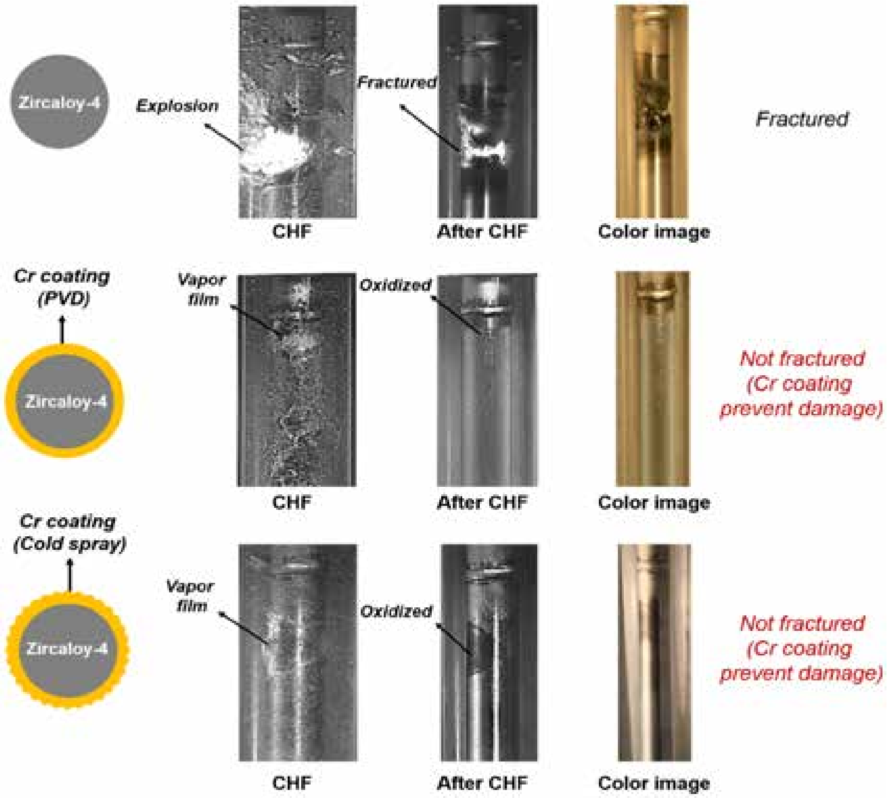


Figure 3. Selected results from MIT-led project: High-speed images at and after CHF for the bare Zr-4 material, Cr-coated Zr-4 with PVD coating, and Cr-coated Zr-4 with cold-spray coating. Both Cr-coated specimens show excellent resilience during the high-temperature CHF condition compared to bare which induces material failure at CHF.

effects including surface wettability, capillary wicking, and the solid-liquid heat conduction, and 2) the heat convection effects resulting from mass flux and liquid subcooling. The effect of cladding material on saturated pool boiling CHF is the most pronounced in pool boiling conditions, since convection heat transfer plays a weak role in these conditions relative to other boiling conditions. Both the onset of nucleate boiling (ONB) and heat transfer coefficient of nucleate boiling (HTC-NB) also show similar tendencies of material dependence with respect to mass flux and inlet subcooling. It also indicates that the formation of oxide layer can influence CHF, ONB, and HTC-NB in the weak forced convection regimes due to the surface morphological factors and the variation of thermomechanical properties.

Project 17-13019

Title: Evaluation of Accident Tolerant Fuels Surface Characteristics in Critical Heat Flux Performance

Investigators: S. Bilbao y Leon (currently NEA); J. Rojas, Virginia Commonwealth University; M. Anderson, University of Wisconsin-Madison; R. Rebak, General Electric; R. Martin, BWX Technologies; R. Harne, Framatome

Selected Highlights: The objective of this project is to perform separate-effects tests of several of the ATF

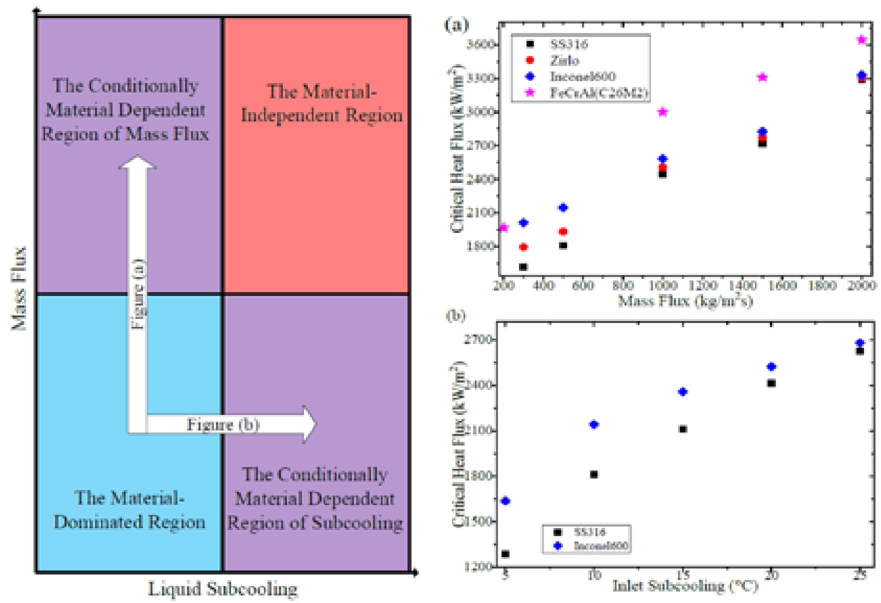
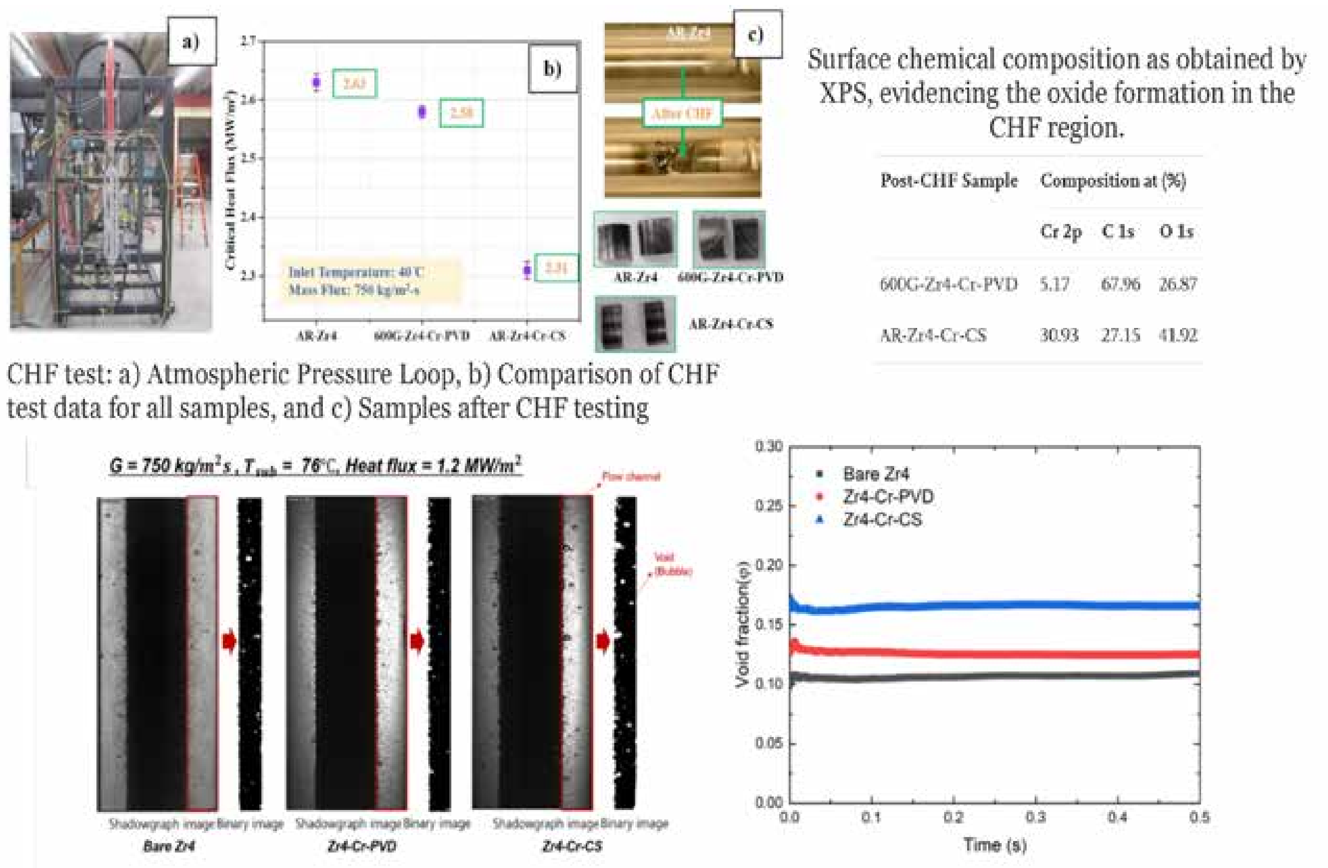


Figure 4. Selected results from UNM-led project: Summary of material-dependent CHF with respect to different flow boiling conditions including selected results. Inlet subcooling is 10 °C for subfigure (a) and the mass flux is 500 kg/m2s for subfigure (b).

concepts to investigate the impact of cladding surface characteristics in CHF under normal and anticipated off-normal conditions. Figure 5 presents selected results of surface effects studies and flow boiling studies as well as void fraction measurements. This project also aims at investigating the evolution of the surface characteristics and surface chemistry under testing conditions. The team evidenced the effect of surface roughness and wettability on critical heat flux (CHF) through the CHF tests in the UW-flow boiling facility under atmospheric pressure. Some results are shown in Figure 6 shows for surface effects studies along with an overview of flow boiling test facilities and results.

The cladding surface temperature was precisely evaluated using a high spatial/temporal resolution optical-fiber (~2.5 mm, ~100 Hz) and the results show that most temperature peaks occurred at 80~95% of the total length (~470mm) on ATF materials. CHF was slightly reduced as wettability decreases for the ATF materials in flow boiling heat transfer under atmospheric pressure. This is thought to be caused by a lower liquid supply on the cladding due to the lower wetting characteristics of the surface, resulting in the early-formation of vapor film which induces CHF. The change in wetting characteristics was mainly seen on Cr-coated Zr-4 by PVD where a decrease of ~10% over bare Zr-4 was



Surface chemical composition as obtained by XPS, evidencing the oxide formation in the CHF region.

Post-CHF Sample	Composition at (%)		
	Cr 2p	C 1s	O 1s
600G-Zr4-Cr-PVD	5.17	67.96	26.87
AR-Zr4-Cr-CS	30.93	27.15	41.92

CHF test: a) Atmospheric Pressure Loop, b) Comparison of CHF test data for all samples, and c) Samples after CHF testing

observed. The surface parameters and surface chemistry after flow boiling critical heat flux tests, conducted at atmospheric pressure, evidenced a slight increase in surface roughness with notable changes in wettability associated with the formation of surface oxides and adsorbed carbonaceous species. CHF on ATF materials FeCrAl alloys (APMT, C26M)) were similar to bare Zr-4 within the 10% deviation. Currently, surface chemistry is being investigated.

References:

[1.] INL Report, edited by G. Su, M. Bucci, and P. Sabharwall, "Investigations on Thermal-hydraulic Behavior of Accident Tolerant Fuel Cladding Materials," INL/EXT-19-56455, January 2020.

Figure 5. Selected results from VCU-led project: The top part of the figure overviews flow boiling and surface characterization studies for CHF at atmospheric pressure, accompanied by surface characteristics and surface chemistry of the tested Cr-coated Zr-4. Bottom shows examples of void fraction measurement on bare Zr4, Cr coated Zr4 with PVD coating, and Zr4 with CS coating heater rod at a heat flux of 1.20 MW/m2 with a mass flux of 750 kg/m2s and subcooling of 76oC. The results clearly show the enhanced void fraction on the rougher surface (Cr coated Zr4 with CS coating) resulting in HTC improvement.

2.2 LWRs FUEL DEVELOPMENT

Crystal Structure Characterization of U-Si Accident Tolerant Fuels

Principal Investigator: Sven C. Vogel
Team Members/ Collaborators: Tashiema L Ulrich, Joshua T. White, A. David R. Andersson and Theodore M. Besmann

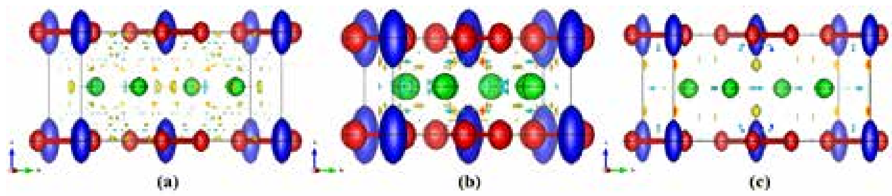


Figure 1. Visualization of the refined crystal structure of U_3Si_{2+x} overlaid with the difference Fourier maps at room temperature (a), 1273 K (b) and room temperature after heating (c). U1 sites are blue, U2 are green and Si are red.

Datasets needed as inputs for silicide-based fuel performance codes have been provided through neutron diffraction experiments and DFT calculations.

Decades of research and operational experience have produced an extensive database supporting the performance of oxide fuel during normal power operations and during postulated accident conditions. Achieving the goal of developing advanced fuel concepts that meet the Department of Energy (DOE) objectives of being robust, demonstrating high performance, and are more tolerant of accident conditions than current fuel systems will require a thorough understanding of the intrinsic properties of the constituent materials. Unlike oxide fuel, the knowledge on non-oxide fuel concepts such as uranium silicide contains gaps and contradicting information. Specifically, there are concerns regarding the stability of the crystal structure for the uranium sesquisilicde (U_3Si_2) phase, which is of interest as a component for non-oxide accident tolerant fuel concept.

Project Description:

Understanding the crystal structure of phases in the U-Si system, in particular the U_3Si_2 structure, is important as it influences fuel properties such as fission product location and transport, thermal expansion anisotropy, defect behavior, radiation damage, and fuel-cladding interactions, among others. The goal of the project is to provide the datasets needed for fuel performance codes that model the aforementioned properties. The experimental objectives of the project are: 1) investigate the crystal structure of U_3Si_2 and other compounds as a function of temperature to provide datasets for the lattice parameters, anisotropic atomic displacement parameters, defects (interstitial atoms), and atomic positions as a function of temperature, 2) provide information regarding the solubility limit as a function of temperature and composition by testing stoichiometric U_3Si_2 as well as hyperstoichiometric U_3Si_{2+x}. The objective of this research is to use density functional theory (DFT) calculations to predict the uranium and silicon defects and their formation mechanisms and validate against experimental data collected by neutron diffraction. The data obtained from the successful completion of this project can directly be used in fuel performance modeling to assess the performance of silicide based fuel compared to the current fuel system.

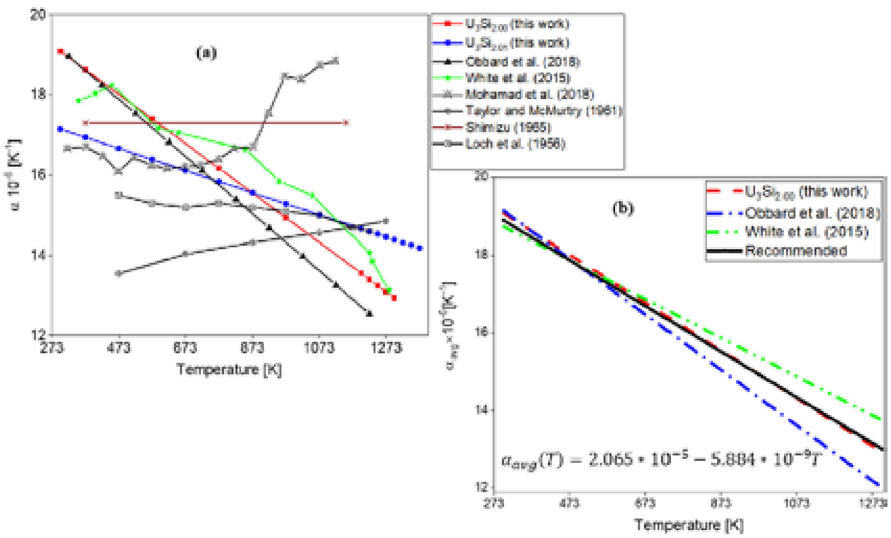


Figure 2. a) Average CTE for U_3Si_2 (red) and U_3Si_{2+x} (blue) compared reported values. (b) Recommended average CTE foe U_3Si_2.

Accomplishments:

The objective of the project is to provide datasets such as lattice parameters, thermal expansion, composition limit and defect chemistry needed as input for fuel performance codes through experimental and theoretical research. Samples were fabricated by Joshua White at the Los Alamos Fuels Research Laboratory and neutron diffraction data of the U_3Si_2 and USi compounds in the U-Si system were collected the at Los Alamos Neutron Science Center (LANSCE) by Sven Vogel. The analysis of the data was done by Tashiema Ulrich and Sven Vogel while Tashiema Ulrich was under the advisement of Theodore Besmann at the University of South Carolina. The diffraction data analysis found that the crystal structure in U_3Si_2 (tetragonal P4/mbm space group) is stable up to the maximum temperature investigated, 1373 K (see Figure 1 for visualizations of the structure included anisotropic thermal motion). An equation for the coefficient of thermal expansion (CTE) as a function of temperature was developed using the results from this work and the assessed literature (see Figure 2). DFT calculations by David Andersson were used to predict that a low defect formation energy and a high entropy for Si interstitial defects give rise to Si-rich non-stoichiometry in U_3Si_2 at elevated temperatures, in agreement with the experimental findings. The enthalpy of formation along with the entropy of each point defect were used to calculate the non-stoichiometry in U_3Si_2 providing information on solubility limit. These results provide experimental evidence that hyperstoichiometric U_3Si_{2+x} exists at temperatures as low as ambient condition, whereas previous modelling predicted that U_3Si_2 is a line compound and excess Si would lead to precipitation of additional phases. The latter would be potentially detrimental to the mechanical integrity of fuel pellets due to different thermal expansion coefficients of different U-Si phases causing mechanical stresses.

Demonstration and Property Measurements on Ceramic Test Articles for Accelerated Fuel Qualification

Principal Investigator: Joshua T. White
Team Members/ Collaborators: David Frazer, Najeb Abdul Jabbar, Chris Grote and Tarik Saleh

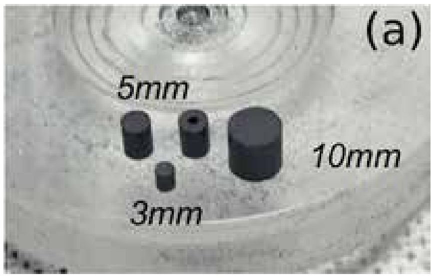
Small scale test specimens as well as properties measurements have been demonstrated providing a path toward accelerated fuel qualification of nuclear fuels.

Interest exists with the Advanced Fuels Campaign (AFC) and industry partners to accelerate the qualification of nuclear fuels. One issue with qualification is conducting irradiations, which can take considerable time to reach the desired burn ups to qualify a given fuel. To circumvent the time required, two different styles of irradiations have been proposed within the AFC. The first, FAST in the Advanced Test Reactor at Idaho National Laboratory (INL), is a scaled down geometry of a reference reactor pellet design, where the enrichment is increased to maintain linear heat generation rates across the radius of the pellet. MiniFuel irradiations have also been proposed at the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory (ORNL), which limits the pellet geometry to ~3mm diameter by 0.3mm thick to facilitate separate effects style irradiation tests [1]. The design for both irradiations is anticipated to greatly accelerate irradiations and also limit sample volumes to accelerate post irradiation examination by decreasing the dose from the irradiated fuel.

Project Description:

The above accelerated irradiation test configurations require the ability to fabricate new pellet geometries. The current work at Los Alamos National Laboratory (LANL) focused on addressing two issues: 1) extending the range of geometries for the FAST irradiations as well as demonstrating the ability to densify accident tolerant fuel samples, and 2) conduct studies on the methods to measure the thermophysical properties on small scale test specimens. The former focused on sintering reference accident tolerant fuel candidates, including large grain sized UO_2 as well as UN. In the case of large grain sized UO_2, sintering profiles have been determined to sinter undoped UO_2 microstructures to grain sizes representative to industry Accident Tolerant Fuel (ATF) doped UO_2, which is on the order of 30 µm as compared to the reference undoped case of ~10-15 µm. Small scale geometries were also expanded in FY20 to include geometries down to 3 mm as well as smaller annular geometries to demonstrate FAST advanced reactor ceramic geometries.

The focus for the determination of thermophysical properties on small scale test specimens was directed towards building an experimental framework to measure the thermal diffusivity and high temperature mechanical behavior. This required the installation of a new light flash analysis instrument in the fuels research laboratory (FRL), which is equipped with a Xe lamp to shorten the light pulse width within the theoretical sample thickness range of interest for MiniFuel samples. Alongside, this new flash analysis capability, mechanical deformation experiments (via nanoindentation)



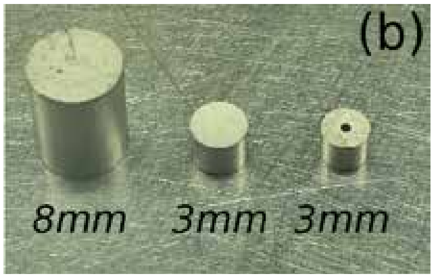


Figure 1. Example FAST geometry pellets for (a) UO_2 and (b) UN. The UO_2 pellets are 90% dense, while the UN pellets achieved 92.5% of the theoretical density of the material.

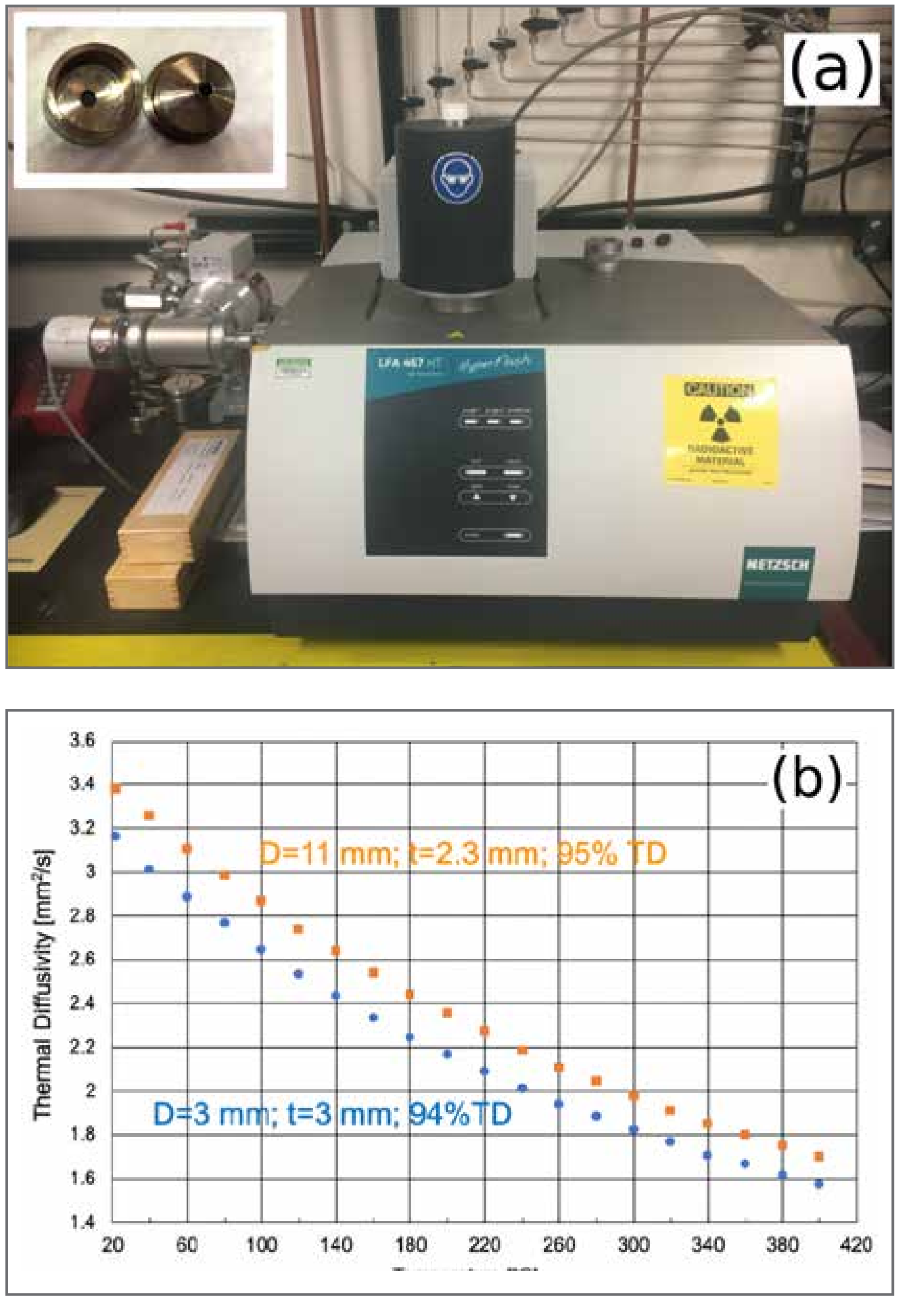
at elevated temperature of small specimens have been conducted using a Hysitron Triboindenter outfitted with a XSOL 800 hot stage. Measurement of the mechanical properties is limited to 800°C and evaluated dislocation flow on (U,Ce) O_2 as a surrogate for mixed oxide fuel.

Accomplishments:

The first aspect of the research was to demonstrate the ability to fabricate new geometries for FAST irradiations as well as sinter different accident tolerant fuel concepts of interest. An example of geometries is shown in Figure 1 (a-b), which highlights the both UO_2 as well as UN pellets. Pellets ranging from reference 10 mm down to 3 mm diameter have been achieved on UO_2 to 90-97% of the theoretical density, while UN pellets were densified to 92-95% density. Pellets down to the size of 3 mm by 3 mm have been demonstrated along with annular variants with a 1 mm hole in the center, shown in Figure 2 (b). Of note is that the smaller geometries tend to have lower densities compared to the larger monolithic variants, which is related to difficulties in controlling the pressing pressures at these small geometries. Defects were also observed in some of the annular pellets resulting from the ejection of the central punch. Progress towards large grain sized UO_2 was also achieved yielding undoped pellets with grains on the order of 30 μm, similar to the doped UO_2 ATF grain sizes produced in industry. UO_2 pellets with FAST geometries and large grain sizes will be demonstrated in a milestone due at the end of this FY to allow for future irradiation tests to evaluate the fission gas release (FGR) effects with enhanced grain sized UO_2, elucidating the effects that defective (doped) UO_2 has on FGR.

Progress towards measuring thermophysical properties was also advanced in the last fiscal year. The installation of an LFA467 HyperFlash instrument was accomplished. This unit has a Xe flash bulb with a light pulse that is 10 times faster than a standard laser unit. This facilitates the measurement of thin samples, which was demonstrated on UO_2 and U_3Si_2

Figure 2. The installed LFA467 HyperFlash system in the Fuels Research Laboratory at LANL in (a). Data collected on a 3 mm MiniFuel and 11mm reference UO_2 sample, showing general trends in the thermal diffusivity between the two geometries. The inset in (a) shows the MiniFuel sample holder that was designed for the LFA.



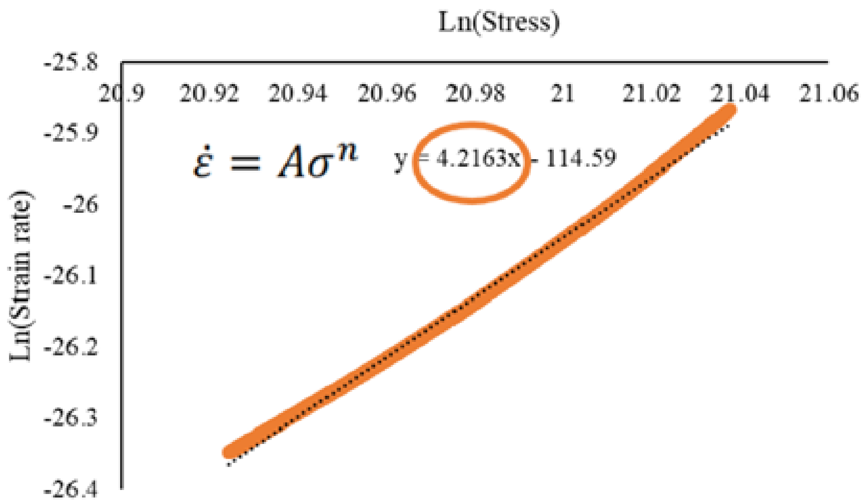


Figure 3. Example nanoindentation creep experiment on (U70,Ce30) O_2 specimen using an in-situ heater at 800 °C. The value circled is the stress exponent of the compound.

samples that were 3 mm in diameter by 3 mm thick. The representative UO_2 data and reference 10 mm UO_2 thermal diffusivity data are shown in Figure 2. The UO_2 thermal diffusivity data is within 6% of the reference data, while the U_3Si_2 was within 10-15% of the reference values. The uncertainty is largely tied to the carbon coating, which is required to improve energy absorption and surface emissivity. While normally a negligible effect for standard sized pellets, the smaller geometries will require corrections to account for the contributions to diffusivity from the coating. Nanoindentation creep measurements were also performed on $(U,Ce)O_2$ in a high temperature in-situ nanoindenter stage to 800°C. Results for this are shown in Figure 3, which shows the natural log of the strain rate as a function of the natural log of the stress. During stead state creep, an exponential behavior is expected, allowing extraction of the stress exponent. This was determined for multiple concentrations CeO_2 in $(U,Ce)O_2$. Stress exponents ranged from 5-7, which is within the dislocation flow regime at 800°C. Both measurements techniques will provide methods to extract valuable data on irradiated samples after accelerated test irradiations.

References:

[1.] Petrie, C.M., Burns, J.R., Raftery, A.M., Nelson, A.T., Terrani, K.A., Journal of Nuclear Materials, 526 (2019) 151783.

Postirradiation Examination of First Generation MiniFuel Samples

Principal Investigator: Jason Harp
Team Members/ Collaborators: Robert Morris, Christian Petrie, Joseph Burns, Darren Skitt and Kurt Terrani

The first postirradiation examination (PIE) results on uranium nitride (UN) kernels and UN coated particles from a MiniFuel irradiation are now available. The MiniFuel irradiation vehicle provides a means to quickly accumulate burnup in nuclear fuel while maintaining the fuel at a steady temperature condition. In this initial irradiation, the accumulated burnup was modest, and the fuel demonstrated good performance with minimal swelling and fission gas release. The initial PIE also validated the modeling and simulation tools used to predict conditions in MiniFuel capsules.

Project Description:

Historically, the development and qualification of novel nuclear fuel concepts, and even incremental changes to established nuclear fuel concepts, have been evaluated by iteratively irradiating fuels in prototypic fuel geometries, neutron flux conditions, and temperatures in integral fuel tests that approximated the fuel concept environment. During iterations, conditions were made incrementally more strenuous until fuel failure was observed [1]. Renewed interest in novel fuel concepts now demands a more agile approach to irradiation testing. The evaluation of new or minimally studied fuel concepts requires accelerated separate effects testing to rapidly screen fuel concepts at several different irradiation conditions while leveraging modeling and simulation capabilities to target the experimental evaluations of key fuel properties that have the greatest impact on fuel performance [2]. The MiniFuel irradiation vehicle is designed to rapidly evaluate nuclear fuel under well-controlled conditions while minimizing the number of variables that occur during irradiation [3]. In MiniFuel irradiations, heating is decoupled from the fission rate so that concepts can be evaluated under a specific condition—such as temperature, composition, and geometry—while simultaneously varying other conditions.

Postirradiation examination of nuclear fuel was also historically done on a large number of fuel pins that were evaluated at the macroscopic scale to

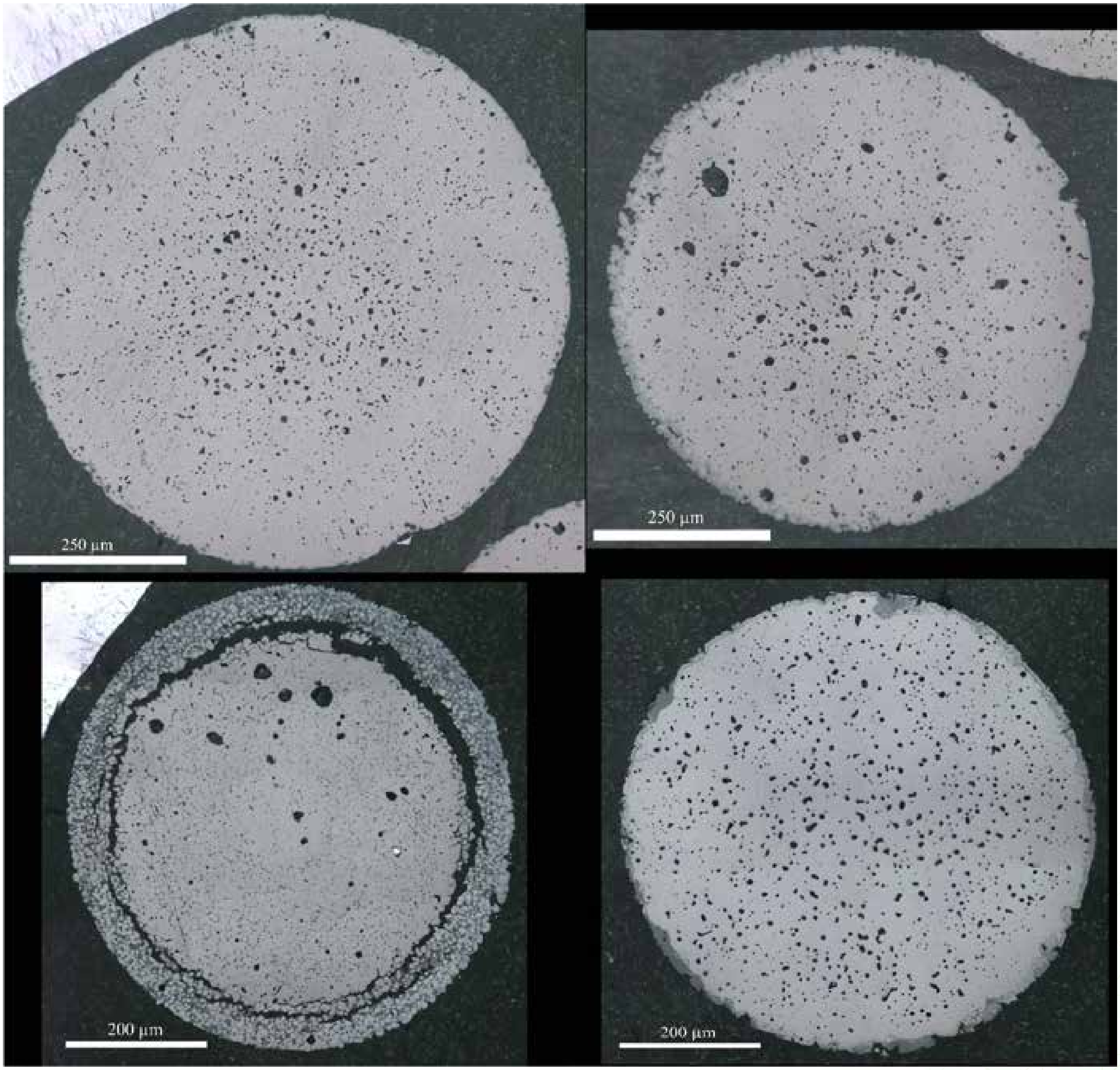


Figure 1. Postirradiation examination of first generation MiniFuel samples.

MiniFuel has been demonstrated through fabrication, irradiation and now PIE to be a valuable tool for screening novel fuel concepts and performing separate effects irradiation testing.

detect failure and large deformation of the fuel geometry. The evaluation of fuel by PIE has become more advanced with more techniques applied to irradiated fuel to better understand the phenomenological mechanisms behind fuel failure. However, performing complex materials science on a large number of integra fuel tests is often impractical. Separate effects test, like MiniFuel, allow PIE to focus on a few variables and more quickly evaluate fuel performance. This approach to testing allows for the developing or improving fuel performance models used in fuel performance codes like BISON.

Accomplishments:

The irradiation performance of the first MiniFuel irradiation test and samples was evaluated through detailed PIE. The first irradiated samples were UN kernels with several different chemistry variants and UN coated particles (UN TRISO). Initial visual examinations revealed no noticeable irradiation induced changes in the kernels or particles. The fission gas release for this fuel is generally in line with athermal recoil release as expected. The microstructural examination of these kernels revealed good performance for these irradiation conditions. Burnup in the fuel was evaluated by a variety of methods. Gamma spectrometry was used to evaluate burnup in all the kernels and particles, and mass spectrometry was used to evaluate burnup for the UN kernels. Very few irradiation-induced changes can be observed in the microstructure. An example of the microstructure of 4 of the kernels from first MiniFuel irradiation is shown in Figure 1. Electron microscopy was also used to evaluate the microstructure and secondary phases observed in the optical microscopy. All the phases detected at this burnup were related to fabrication artifacts not irradiation induced phases. These positive results justify the continuation of ongoing High Flux Isotope Reactor (HFIR) irradiations of UN kernels and UN TRISO to higher burnups.

More generally this PIE shows how the MiniFuel design can be used for the rapid screening of novel fuel concepts. Because the MiniFuel experiments are significantly different than traditional fueled experiments, this PIE also served as a shakedown test of the PIE techniques that will be applied to future MiniFuel experiments. The predicted temperatures for the fuel (approximately 500°C) closely matched the measured temperatures (approximately 450°C) derived from temperature monitors (TM) in the capsules. The measured Burnup in each capsule (5.2 to 8.9 MWd/kgU) was within 10% relative error with the burnup predicted from neutronics simulations. Experimental evaluations of burnup and temperature provided valuable feedback for simulations of MiniFuel targets that are currently being designed and fabricated for insertion into HFIR.

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[3.] C.M. Petrie, J.R. Burns, A.M. Raftery, A.T. Nelson, K.A. Terrani, Separate effects irradiation testing of miniature fuel specimens, J. Nucl. Mater. 526 (2019) 151783. doi:10.1016/j.jnucmat.2019.151783.

Development of Accident Tolerant Oxide Fuel Grain Growth Kinetic Models

Principal Investigator: Tashiema L Ulrich
Team Members/ Collaborators: David M Frazer, Joshua T. White and Tarik A. Saleh

Averaged grain size of 36.4 ± 15.1 μm have being attained from sintering in CO_2 gas for 24 hours, which is comparable to grain sizes that are achieved by commercial processes.

For the last six decades, industry research teams have demonstrated that the addition of oxide dopants to UO_2 fuel during fabrication provides significant improvement to the microstructural properties, specifically a fivefold increase of the grain size [1-2]. Enhanced grain sizes have been shown to improve fission gas release (FGR) within the fuel rodlet, as well as affect creep and hardness of the fuel pellets. The Advanced Fuels Campaign (AFC) is supporting research that seeks to understand the role that dopants have on FGR, which is important for extending the lifecycle of the fuel. It has been suggested that dopants increase not only the grain size, but also enhance fission gas diffusion, potentially negating the benefits of large grains [3]. Since large grain size UO_2 has predominantly been obtained by the addition of dopants, this study has been proposed to grow large grain sized UO_2 without the aid of dopants allowing for studies which decouple the grain size effect on FGR in future irradiation studies.

Project Description:

For the last 50 years there have been ongoing efforts to understand how dopants affect fuel properties such as thermal diffusivity, fission gas release, and creep strength; however, it is challenging to decouple the effects of dopants from effects of large grain size. Decades of experience with UO_2 fuel have yielded a wealth of information regarding the microstructural evolution and sintering kinetics of undoped UO_2 providing critical insight into the atmospheres and temperatures required to yield dense pellets. Unfortunately, many of these studies provide inconsistent or incomplete results on the grain sizes of sintered pellets, which requires further investigations. This project will expand on previous sintering kinetics studies to provide insight into controlling sintering dwell time, sintering atmosphere and stoichiometry of UO_2 to emulate microstructures from Accident Tolerant Fuel (ATF)-derived Cr and/or Al doped fuel commercially produced by industry. Successful completion of the project would allow for a consistent

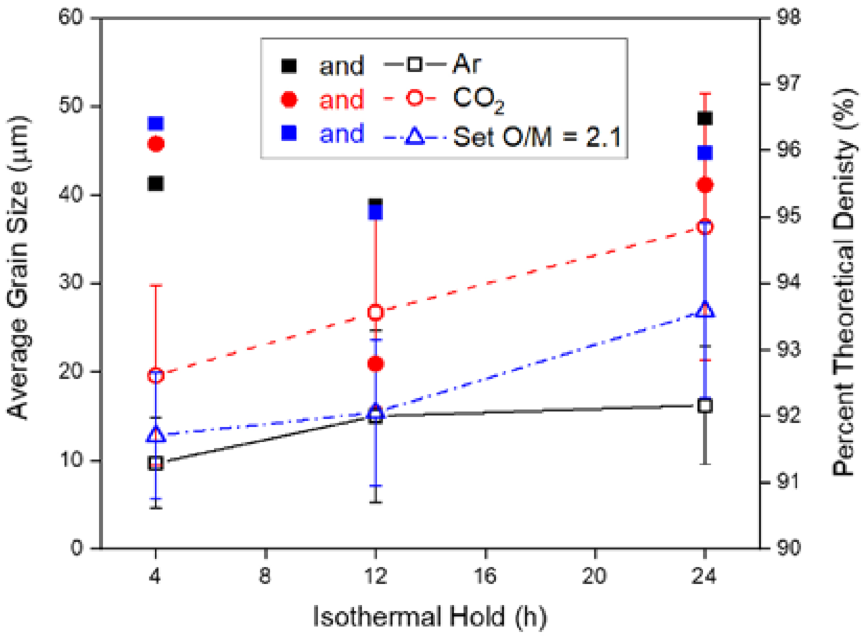

Figure 1. The effect of sintering time and atmosphere on grain size in undoped fuel.

fabrication of large grained undoped UO_2 fuel that could serve as reference fuels for fission gas behavior studies in enhanced UO_2 accident tolerant fuel. Such experiments would provide the data necessary for decoupling dopant effects vs microstructural effects (i.e., large grains) and further enhance the understanding of the mechanisms that control grain growth in UO_2.

Accomplishments:

Sintering studies were performed in a dilatometer to investigate the densification behavior in a variety of gas conditions, ramp rates, and holds. The same instrument profiles were conducting in an STA to monitor the stoichiometry during the sintering schedule to inform the role that crystalline defects have on sintering kinetics in UO_2. The initial UO_2 feedstock had an oxygen to metal (O/M) ratio of 2.16 and a pressing pressure of 200 MPa was used to prepare green pellets.

A 4 hour sintering time at 1550 °C in an inert (Ar) or oxidizing (CO_2) atmosphere with a 5 °C/minute heating/cooling rate produced pellets with an average density of 94%

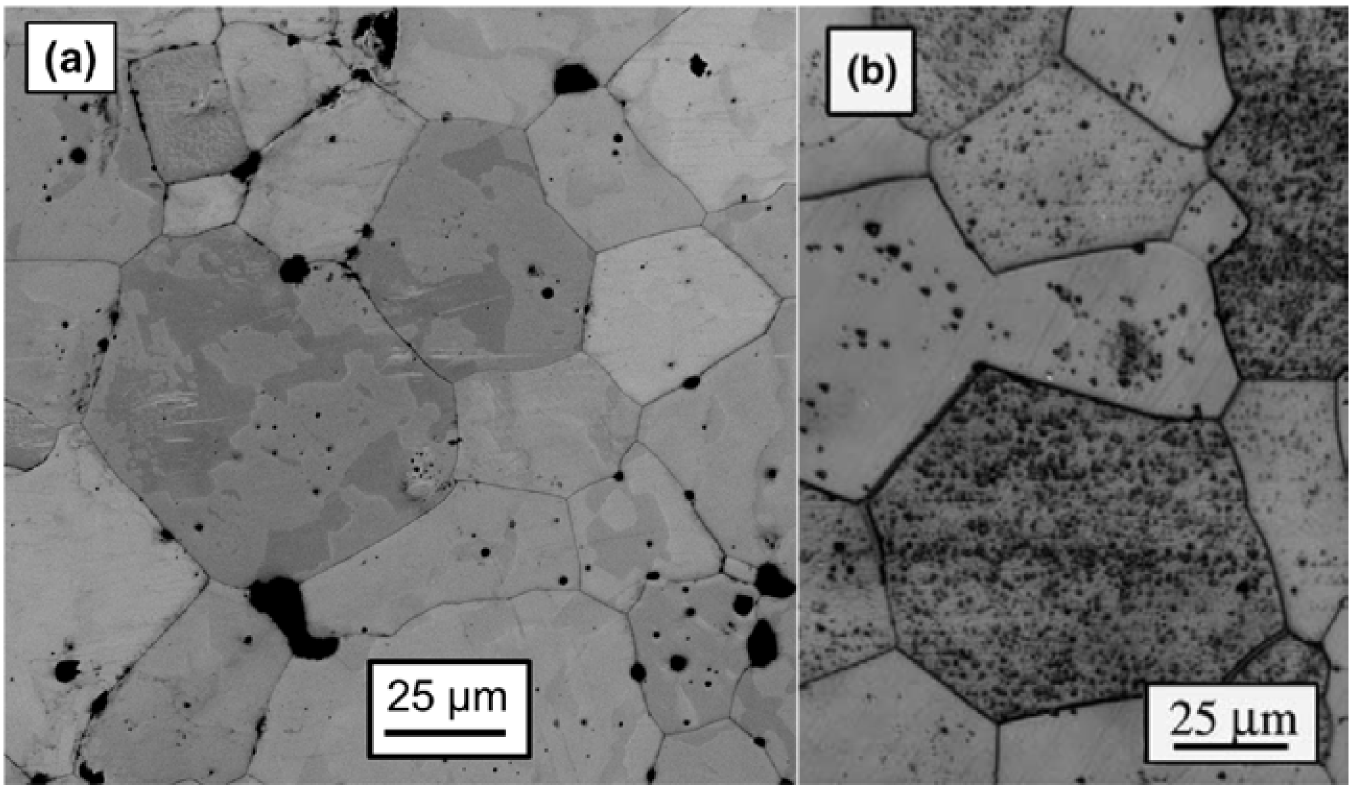

Figure 2. Comparison of between an undoped UO_2 sintered for 12 hours under CO_2 (a) with a 0.05 wt% Cr_2O_3 doped UO_2 sintered under H_2 + 1 vol% H_2O from [2] (b).

theoretical density (%TD) and grain sizes ranging between 9.7-19.6 µm depending on the sintering atmosphere. The standard average grain size in light water reactor fuel is between 10-12 µm [1] and the sintering temperature ranges between 1700 °C to 1800°C in a slightly oxidizing atmosphere. It is understood in the literature that an oxidizing atmosphere introduces defects into the UO_2 crystalline lattice, thus improving grain growth [4]. To that end, CO_2 was chosen as the oxidizing atmosphere. The pellet sintered under CO_2 for 4 hours has an average grain size of 19.6 ± 10.2 µm, which is comparable to that of TiO_2 doped UO_2 fuel (~18 ± 17 µm) [5-6]. The sinterability of UO_2 compacts has been found to depend upon four main factors: sintering time, the characteristics of the starting UO_2 powder, time sintering temperature and the sintering atmosphere. In this study it has been shown that an increase in sintering time can stimulate grain growth, which can be further enhanced by an oxidizing atmosphere. Figure 1 summaries the effect of sintering time and atmosphere on grain growth in undoped fuel. The increase in sintering time from 12 to 24 hours did not produce significant improvement in grain size for samples under Ar atmosphere, however there was improvement in the size from 4 to 12 hours. The opposite effect is true for samples sintered under a fixed O/M ratio of 2.1. The average grain size of 19.2 μm ± 10.2 μm, for the 4 hours CO_2 is similar to those obtained for Cr_2O_3 doped UO_2 pellets fabricated at 1525 °C for 4 hours [2]. Furthermore, the sizes at 12 hours and 24 hours are in line with Cr_2O_3 doped UO_2 pellets fabricated at 1625 °C for 4 hours, Figure 2 shows a comparison of between an undoped UO_2 sintered for 12 hours under CO_2 Figure 2(a) with a 0.05 wt% Cr_2O_3 doped UO_2 sintered under H2 + 1 vol% H_2O from [2] Figure 2(b). Averaged grain size of 36.4 ± 15.1 μm have being attained from sintering in CO_2 gas for 24 hours, which is comparable to grain sizes that are achieved by commercial processes. The results from this study will enable the fabrication of undoped UO_2 control test articles for future FGR irradiation studies to enable quantitative comparisons with ATF doped UO_2 commercially produced fuels.

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Sintering high uranium density fuels in a flash

Principal Investigator: Rubens Ingraci Neto
Team Members/ Collaborators: Darrin Byler, Ken McClellan and Erofili Kardoulaki

Uranium dioxide pellets with good mechanical integrity and densities up to 94 %TD were obtained in less than 30 minutes using current controlled flash sintering with alternate electric fields.

Nuclear fuels with high thermal conductivity, and elevated melting point, such as UN, are an alternative to traditional UO_2 that can contribute to the safe operation of the reactor [1, 2]. Furthermore, UN has high uranium density which allows higher burnup and long-life operation [3].However, sintering of UN requires temperatures greater than 2100 °C and sintering times of about 10 h to achieve 95% of its theoretical density (TD) [4]. Furthermore, UN is prone to oxidation and reaction with steam [5, 6, 7]. Recently, UO_2-UN composites with small additions of UN have been considered density fuel alternative due to the oxidation resistance that is expected to be provided by the UO_2 [8, 9]. Although, this composite is still difficult to sinter. Flash sintering (FS), a field assisted sintering technique (FAST) that combines electric fields and currents, has been shown to drastically reduce the sintering times and temperatures of nuclear fuels.

Project Description:

In this study, FS has been investigated for densification of $UO_{2.00}$ + 10 vol% UN composites. Previously, Valdez et al. [10] used FS to obtain $UO_{2.16}$ pellets up to 91 %TD, although, these pellets were severely cracked due to high thermal gradients that developed. During the traditional FS method used by Valdez et al. [10], the electric current flowing through the pellet increased from 0 A to ~ 11 A in a fraction of second. This uncontrolled current surge led to rapid Joule heating and consequent damages to the pellets. In addition, the DC electric field used for the FS experiments induced oxygen vacancy migration and overheating of the cathode causing a geometric distortion of the pellet as well as an O/M gradient along the pellet. These results suggest that controlling the FS parameters is vital to achieve dense pellets with good mechanical integrity. Therefore, the first stage of this study aimed to analyze how the FS parameters, mainly the electric current and electric field waveform, affect the sintering kinetics of $UO_{2.16}$. By controlling these FS parameters, $UO_{2.16}$ pellets with good mechanical integrity and up to 94 %TD were obtained. These FS conditions were then applied to experiments with $UO_{2.00}$+ 10 vol% UN composites.

Accomplishments:

The densification of $UO_{2.16}$ pellets during FS, at controlled current rates, is compared in Figure 1 with that during conventional sintering (CS) at different heating rates (2, 5, and 10 °C min^{-1} up to 1600 °C). For all FS tests, the furnace was heated up to 600 °C. Following this, an AC electric field was applied and the current flowing through the sample was increased linearly in a controlled manner up to 12 A (RMS). As shown in Figure 1, FS at 32 mA mm^{-2} min^{-1} led to 94%TD in 25 min in contrast to 92 %TD after 14 hours of CS at 2 °C min^{-1}.

During controlled current rate AC-FS of $UO_{2.16}$, it was observed that the lowest current rate (32 mA mm^{-2} min^{-1}) led to reactions near the electrodes, seen as porosity in Figure 2(b). Whereas at moderate current rates (126 mA mm^{-2} min^{-1}) pellets with good integrity and densities above 92 %TD were obtained (see Figure 2(c)).

To increase understanding of the sintering kinetics during FS of UO_2 the sintering activation energy of $UO_{2.16}$ was calculated using the master sintering curve (MSC) method [12] as shown in Figure 3. The activation energy for FS, 107.5 kJ mol^{1}, was found to be closer to the activation energy reported for SPS of $UO_{2.16}$ by Chen et al. [13], 140 kJ mol^{-1}, than the activation energy calculated during CS, 380 kJ mol^{-1}. This finding highlights the similarities of the electrical effects present in FS and SPS, and points to a similar sintering mechanism. UN pellets and other UO_2 composites have been successfully sintered before to high densities using SPS [14, 15].

Once the best parameters for controlled current rate AC-FS were identified as a current rate of 126 mA mm^{2} min^{-1} up to 12 A (RMS) and a dwell of 2 min at this maximum current, based on experiments on $UO_{2.16}$, a UO_2-UN composite pellet with 10 vol% UN was then subjected to this same test condition under an Ar+N_2 atmosphere. The pellet density obtained was 75%TD in 14 minutes. It is expected that further densification can be obtained if the current during FS exceeds 12 A, the limit of the current FS setup. By adding UN to the composite, its electrical resistivity decreases forcing the need of higher electrical currents to obtain elevated electric power densities. Additionally, it was found that UN fully transformed to U_2N_3 during FS as shown in Figure 4. This is probably because of N_2 present in the atmosphere when the pellet was ~1300ºC. Therefore, better control of the pellet temperature and atmosphere during FS are required.

To further improve the sintering kinetics of UO_2-UN composites, a new pathway has been conceived by combining axial pressure with the electrical effects of FS. A pressure assisted flash sintering equipment has been designed and is currently being assembled. This innovative sintering method has the potential to deliver high density pellets, with good mechanical integrity and tailored microstructures at low furnace temperatures and in a few minutes.

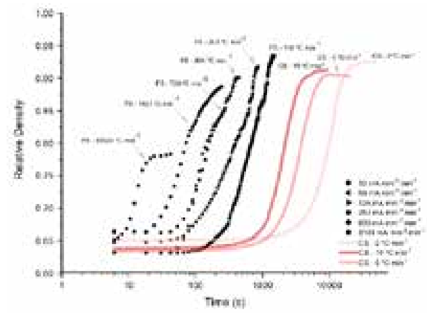

Figure 1. Densification profile of $UO_{2.16}$ during CS at different heating rates and FS at various current rates.

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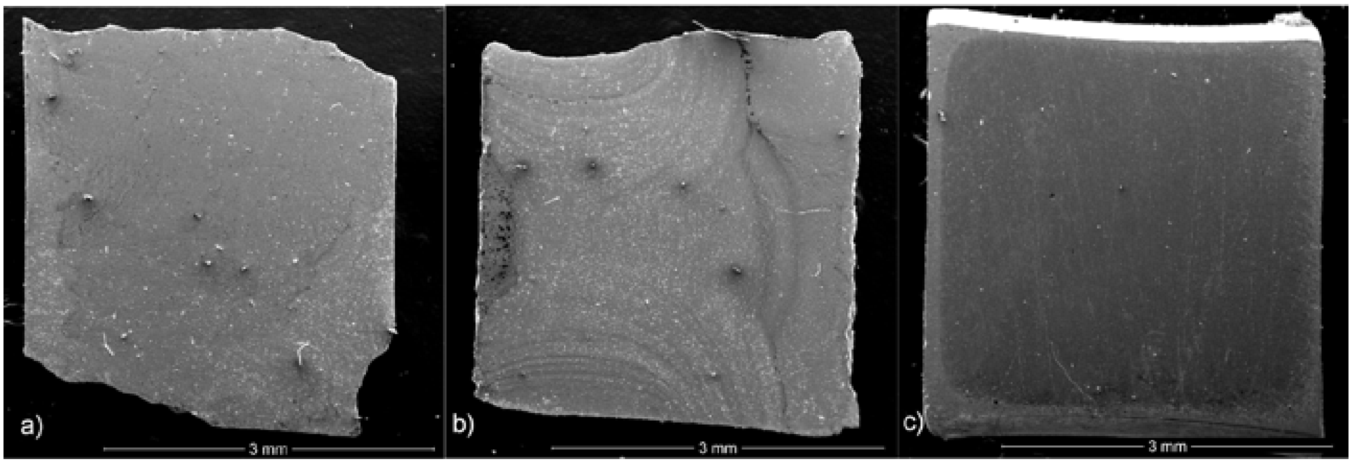

Figure 2. SEM images showing $UO_{2.16}$ pellet microstructures, (a) after CS at 10ºC min^{-1}, (b) after FS at 32 mA mm^2 min^{-1}, and (c) after FS at 126 mA mm^2 min^{-1}

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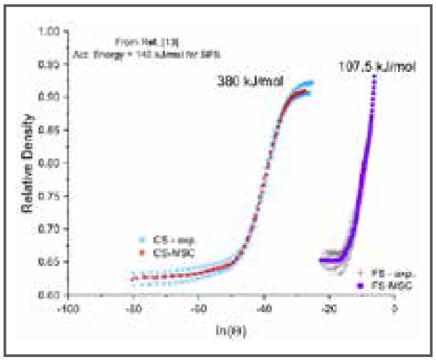

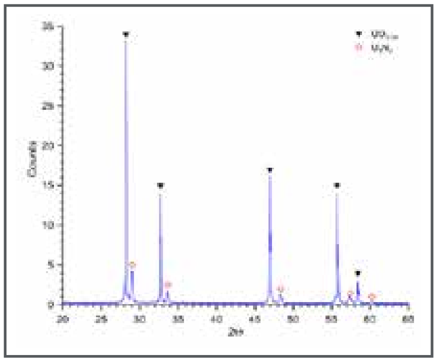

Figure 3. MSC and sintering activation energies of $UO_{2.16}$ for CS and FS.

Figure 4. XRD result of $UO_{2.00}$ with 10 vol% UN pellet after FS showing the formation of uranium sesquinitride.

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2.3 LWR CORE MATERIALS

Progress in development and characterization of coatings for SiC/SiC cladding

Principal Investigator: Peter Mouche (ORNL)
Team Members/ Collaborators: Peter J. Doyle (UTK), Takaaki Koyanagi (ORNL), Stephen S. Raiman (ORNL) and Yutai Katoh (ORNL)

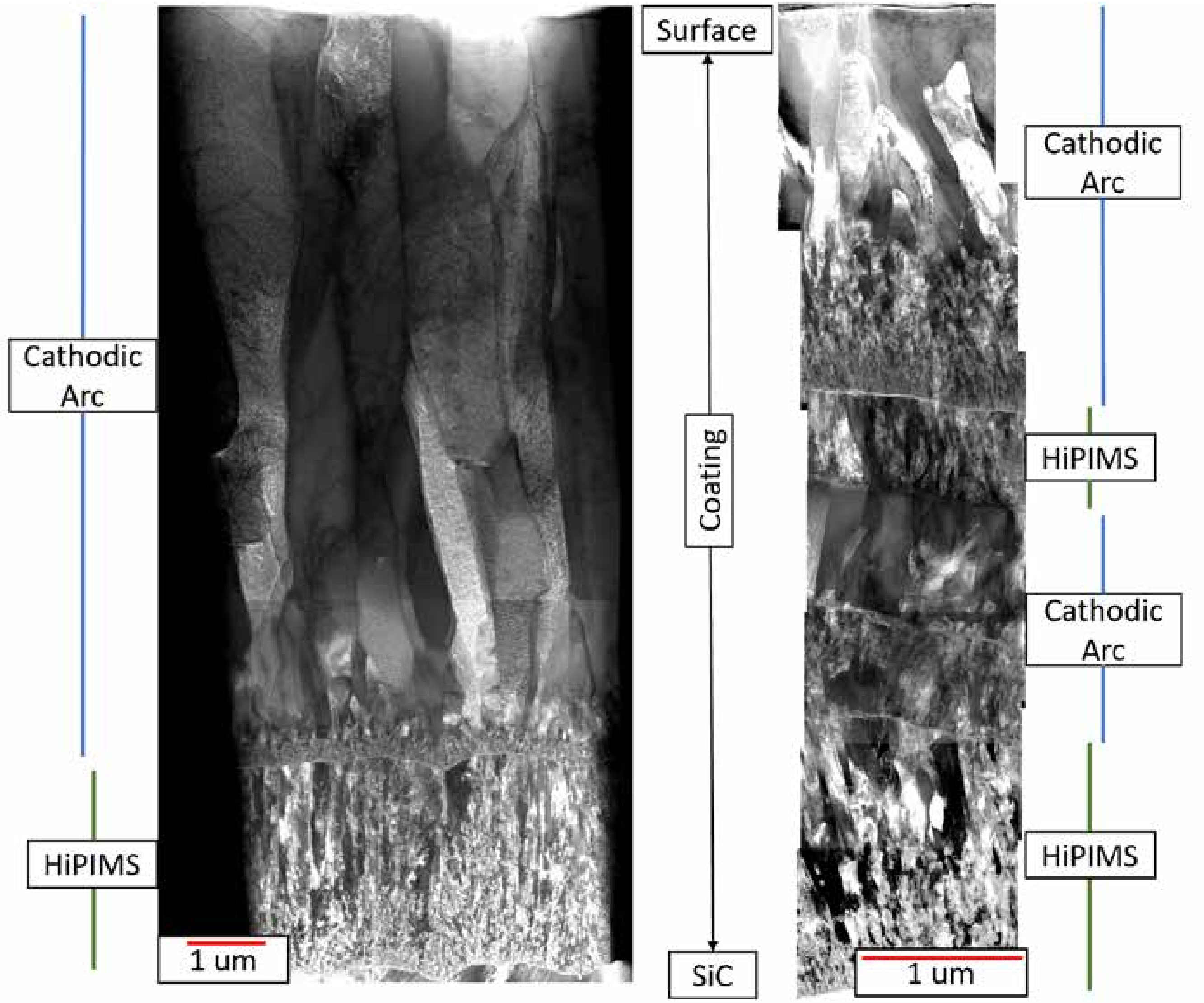
Testing of various coating deposition methods identified the residual stress range at which Cr coatings will remain adherent and a microstructural method to increase crack resistance.

Silicon carbide (SiC) fiber reinforced SiC matrix composites provide many potential benefits as compared to traditional zirconium-based alloys for light water reactor (LWR) fuel cladding. However, several technological barriers still need to be overcome before it can be implemented making it a high-risk, high-reward goal. Hydrothermal corrosion and loss of hermiticity due to crack networking are the two issues targeted by developing coatings on SiC. Previous corrosion and irradiation work at Oak Ridge National Laboratory (ORNL) identified pure Cr as a promising material over other ceramic coatings. The major challenge is solving tensile stress build up in the coating from radiation induced swelling of SiC. This swelling mismatch between the SiC and the coating can lead to cracking or delamination of the coating. Multiple Cr coatings were deposited to investigate pathways to mitigate crack formation in the coating as well as the corrosion response if cracks do form.

Project Description:

The objective of this work is to understand the effects of deposition parameters on the morphology of coatings on SiC, and leverage this to design a coating that is specific to the needs of nuclear. As SiC swells under neutron irradiation the range of stresses a coating can withstand, crack propagation through the coating, and how any cracking will affect corrosion needs to be investigated. By understanding the fundamentals of these processes, a robust coating can be created. This will bring SiC/SiC composites one step closer to being able to be deployed and realizing its potential advantages as a more stable cladding.

Experiments focused on characterizing two main physical vapor deposition (PVD) coating architectures deposited on monolithic SiC. The first were hybrid high power impulse magnetron sputtering (HiPIMS) combined with cathodic arc deposited by Acree Technologies Inc. to test crack resistant coatings with bimodal mechanical properties that are single phase (Figure 1). Specimens with pre-cracked coatings were exposed to various LWR water chemistries for 500 hours to determine if oxide formation could "heal" cracks if they form during service. The second set of coatings were DC magnetron sputtering (DCMS) coatings, deposited at Manchester Metropolitan University (MMU), to determine how much compressive residual stress can be fabricated into coatings to offset the SiC swelling. The coating properties, as measured by transmission electron microscopy (TEM), scanning electron microscopy (SEM), x-ray diffraction (XRD) residual stress measurements, and nano-indentation were related



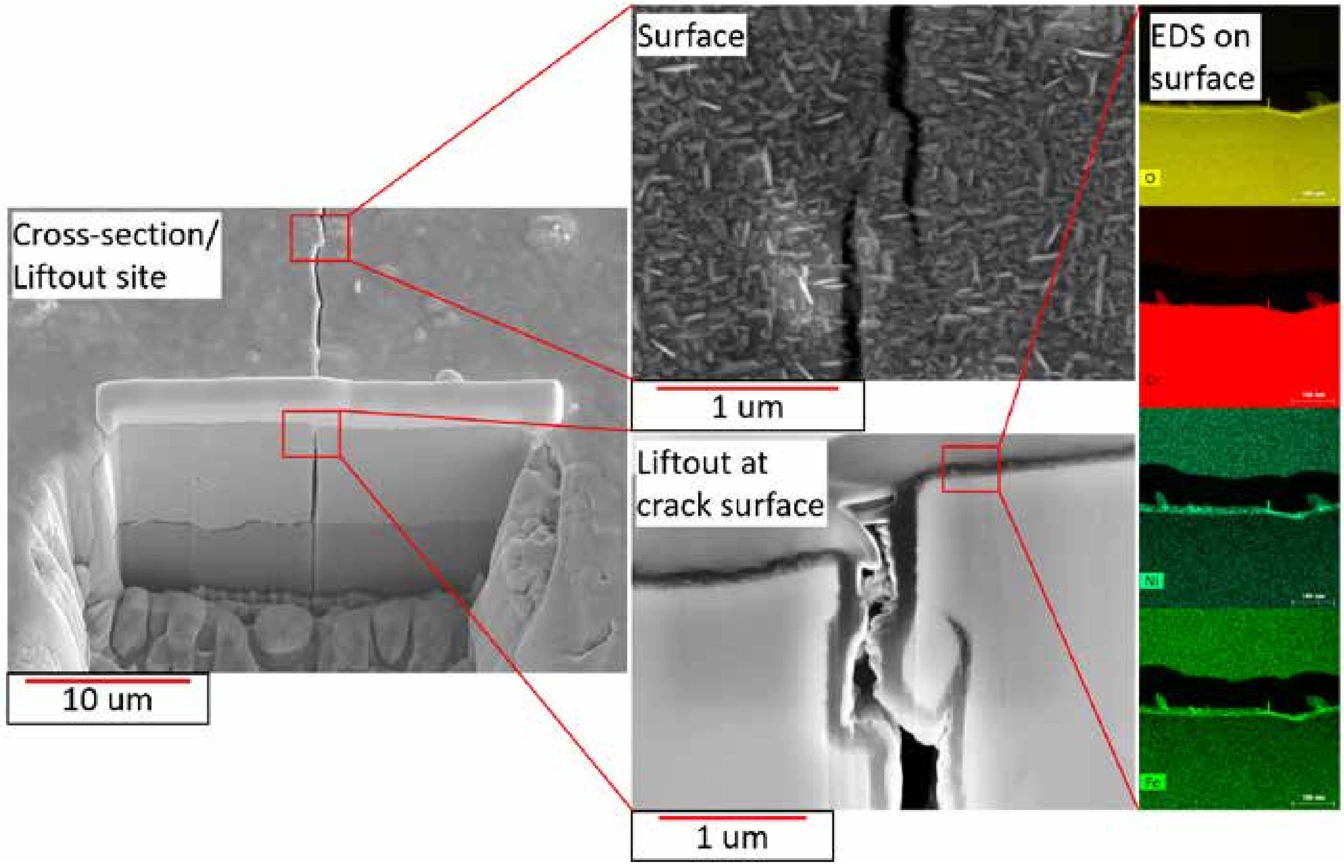
to adhesion test and deposition parameters. All characterization and corrosion were performed at ORNL using the Low Activation Materials Development and Analysis (LAMDA) Laboratory, the Hydrothermal Corrosion Laboratory, as well as other facilities (Figure 2).

Accomplishments:
Investigations into the feasibility of hydrothermal corrosion driven crack healing was completed in fiscal year (FY) 2019. Samples with pre-existing cracks from tensile stresses were exposed to boiling water reactor hydrogen water chemistry for 500 hours by Peter Doyle. Oxide evolution

Figure 1. Hybrid single phase Cr coatings with layers deposited by high power impulse magnetron sputtering and cathodic arc.

Figure 2. Loading Cr coated SiC coupons for an X-ray diffraction residual stress measurement at ORNL.

inside of the cracks was investigated by using focused ion beam milling to both expose crack cross-sections, as well as fabricate liftouts for Scanning Transmission Electron Microscopy (STEM)/ Energy Dispersive X-Ray Spectroscopy (EDS) characterization by Daniel Morral. It was observed that the corrosion rate on both the surface and interior of the crack was too low to form an oxide layer in a reasonable amount of time (Figure 3). Additionally, due to the stress distribution cracks will be wider at the base further hindering this process. Therefore, coatings must be designed which stop the formation of cracks. By changing the kinetic energy of the ions during deposition through sample biasing a compressive residual stress can be imparted into the sample. DCMS coatings produced by Adele Evans at MMU with various levels of compressive residual stress were characterized at ORNL. It was found that 0.8 GPa of compressive stress can be imparted into coatings before adhesion is affected. If compressive stresses become too large the SiC will form microcracks reducing the coating adhesion. Previous work at ORNL showed that tensile stress cracking and spallation initiates above ~0.35 GPa of tensile residual stress so a large stress buffer can be built into the coating. Toughening the coating to either blunt or deflect cracks was the other strategy investigated. Nanoindentation and



microcantilever testing of the HiPIMS and cathodic arc coatings showed that the HiPIMS coatings were very hard with nanoscale grains while cathodic arc coatings were surprisingly ductile with large grains. Initial testing of hybrid HiPIMS and cathodic arc coatings with microindentation found that small cracks could be stopped by the softer cathodic arc chromium, and the interface between the HiPIMS and cathodic arc layers can serve as a weakened pathway for crack deflection.

To test the two different crack mitigation pathways, samples have been fabricated this FY for two different neutron irradiation experiments. Tubes of Chemical Vapor Deposition (CVD) SiC were coated with two different HiPIMS and cathodic arc coating architectures and sent to collaborators for the Il Trovatore irradiation in BR2 at SCK•CEN. At ORNL, coupon samples with 11 different coating architectures were machined from the stock of well characterized DCMS, HiPIMS and cathodic arc coupons along with an ion-assisted magnetron sputtered sample set supplied by the University of Wisconsin. These will be irradiated in High Flux Isotope Reactor (HFIR) at 300°C to target SiC swelling effects.

Figure 3. Representative microscopy images of corrosion in pre-cracks. Scanning electron microscopy show the cross-section, surface, and liftout. Chemical mapping using scanning transmission electron microscopy highlights the oxide thickness.

Wear Performance of a Candidate ATF Cladding Coating in Grid-to-Rod Fretting

Principal Investigator: Jun Qu
Team Members/ Collaborators: Brady Reed, Rick Wang and Roger Y. Lu (WEC)

A candidate Cr-coating for ATF cladding has demonstrated superior wear resistance in grid-to-rod fretting.

In pressurized water reactors (PWRs), water flow induced vibrations cause contact and rubbing between the fuel rods and the supporting grid, a phenomenon known as Grid-to-Rod-Fretting (GTRF) [1]. GTRF may produce progressive wear damage on the cladding leading to subsequent fuel failure. Such an event is termed a "leaker" and may cause premature and expensive reactor shutdown and maintenance. Various accident-tolerant fuel (ATF) [2] concepts are being developed for enhance the resistance of cladding to the high temperature steam. One approach is use materials more oxidation resistant than reference zirconium, most commonly SiC [2,3] or FeCrAl [3]. Another approach is to apply a protective surface treatment, e.g. pre-oxidation, or coating, e.g., CrAl [4], Cr [5], and FeCrAl [6], to the Zr alloy cladding surface. While Cr-coatings have shown good promises for ATF claddings, their fretting wear resistance was little known.

Project Description:

In this study, fretting wear behavior of a candidate Cr-coating for ATF cladding was investigated in a PWR-relevant environment using a unique bench-scale autoclave testing rig at Oak Ridge National Laboratory (ORNL). Test conditions were designed to mimic the environment in an industrial full-assembly reactor core simulator. Results were compared with that of the state-of-the-art Zr alloy cladding and demonstrated that the Cr-coating could significantly improve the cladding's wear resistance when tested against a commercial Zr alloy grid with or without pre-oxidization. In addition, the Cr-coating also reduced the counterface wear on the ZIRLO grid without pre-oxidation, but slightly increased the wear on the pre-oxidized grid. The encouraging testing results and mechanistic understanding of the performance improvement provides scientific insights for further development of ATF claddings to meet Department of Energy (DOE's) objectives of safe, reliable, and economical operation of the nation's current and next generation reactors.

Accomplishments:

Commercial Zr alloy cladding and grid as well as Cr-coated cladding were supplied by Westinghouse Electric Company (WEC). Microstructure characterization (see Figure 1) revealed that the thickness of the Cr-coating is 30-40 μm and has a limited number of voids. The bonding with the substrate appears excellent and is absent of gaps or cracks. Based on Vickers indentation, the Cr-coating has a hardness of 392 HV, about 40% harder than the commercial Zr alloy cladding. A unique autoclave fretting rig (AFIR, as shown in Figure 2) was recently established at ORNL to study the wear resistance of the claddings and grids [7]. AFIR allows grid-to-rod fretting (GTRF) study of actual cladding and grid materials in a pressurized water environment. The AFIR was designed to mimic the

Figure 1. A unique autoclave fretting-impact rig (AFIR) at ORNL. (top-left) The AFIR control unit, power supply, and plumbing; (top-right) The furnace and test chamber behind the panels; (bottom-left) The autoclave vessel with liquid and gas handling system; (bottom-right) The specimen assembly inside the autoclave chamber.

environment in Westinghouse Electric Company (WEC)'s VIPER system, a full-assembly reactor core simulator. AFIR tests were carried out at 204 °C under a pressure of 20-23 bars. The contact force between a pair of cladding-dimple was 0.5±0.1 N, matching the peak cladding-dimple force of 0.5 N in reactor [8]. The fretting frequency was set at 25 Hz and the oscillation stroke was 75±10 μm. Testing was carried out at both 20.5 hours (AFIR-S) and 100 hours (AFIR-L). Figure 3 compares the appearance of claddings and their respective grids after 20 and 100 hour fretting tests in pressurized water at 204 °C. The conventional Zr-Zr pair showed a darker color after the 100-hr test due to a significant increase in oxidization during the longer test, as expected. In contrast, the Cr-coated cladding had little change in appearance after the 100-hr test, which is indicative of superior

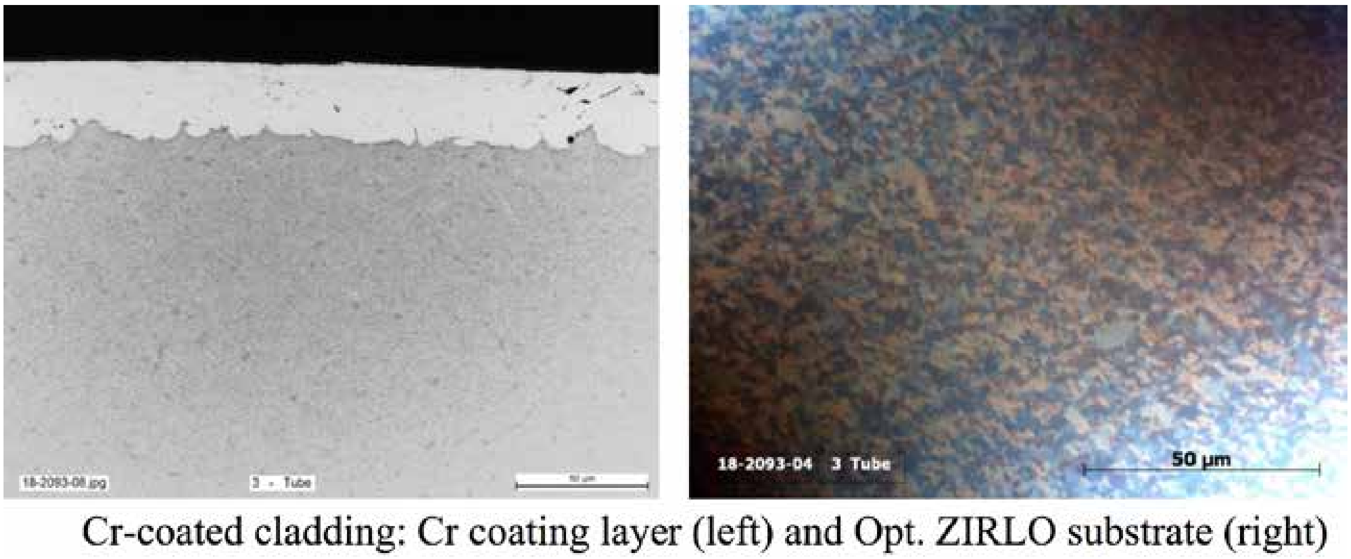


Figure 2. Microstructures of the Cr-coating and Zr alloy cladding.

oxidation resistance. On the other hand, the Zr alloy grid that was tested with the Cr-coated cladding appeared much darker than that against the Zr alloy cladding after the 100-hr test. The accelerated oxidization of the Zr grid is believed to be caused by galvanic corrosion promoted by the large electrode potential differential between Zr (-1.45 V or -2.36 in presence of OH-) and Cr (-0.74 V). Wear results of the long tests (AFIR-L, 100 hours) are plotted and compared with relevant short test results in Figure 4. When tested against the as-received Zr alloy grid, the wear coefficient of the Zr alloy cladding was almost constant for 20 and 100 hours. This suggests a linear progression of the wear loss during the fretting test. In contrast, the wear coefficient of the Cr-coated cladding was 61% lower in the 100-hr test than that in the 20-hr test, indicating a significant decrease in wear loss after running-in. The Cr-coating reduced the volumetric wear by 94% compared to the Zr alloy cladding. Additionally, wear on the counterface (Zr alloy grid) was reduced by 58% when the Cr-coating was used. In addition, grid wear was also reduced when rubbing against the Cr-coated cladding than against the Zr alloy cladding.

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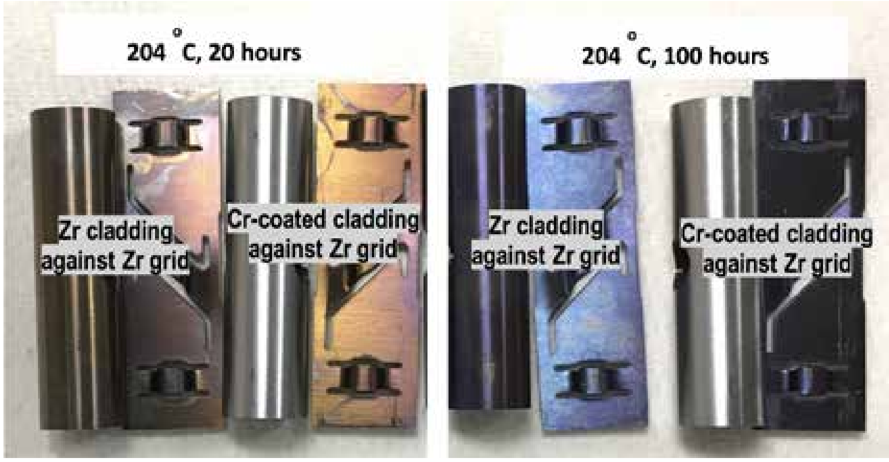


Figure 3. Appearance of the uncoated Zr alloy and Cr-coated claddings after 20 and 100 hours fretting tests in pressurized water at 204 ºC.

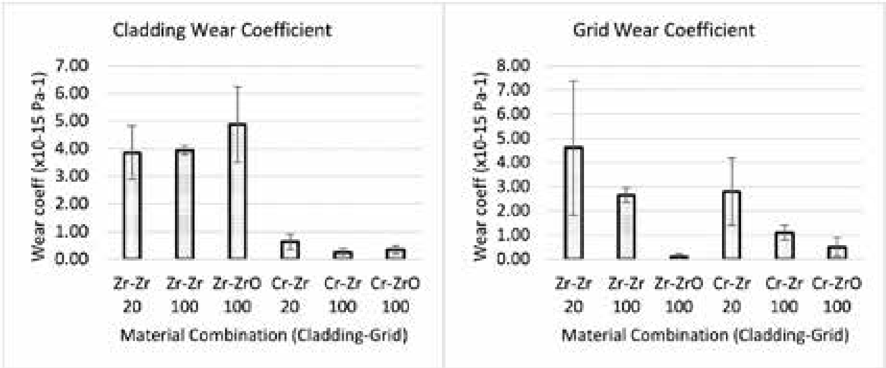


Figure 4. Comparison of cladding and grid wear performance in short (AFIR-S, 20 hours) and long fretting tests (AFIR-L, 100 hours).

Mechanical Testing of FeCrAl Tubing

Principal Investigator: Stuart A. Maloy
Team Members/ Collaborators: Mathew L. Hayne and Carl M. Cady

Tensile properties were measured on C26M tubing providing engineering stress/strain curves from room temperature to 600C.

Because of the events related to the Fukushima reactor incident, innovative new cladding materials are being investigated for light water reactors (LWRs) with improved resistance to accident conditions such as a Loss of Coolant Accident (LOCA). Materials need to be developed and tested to meet these challenging conditions. One possible candidate material for cladding applications is a FeCrAl alloy. Advanced FeCrAl alloys have been developed at Oak Ridge National Laboratory (ORNL). These alloys need to be qualified through mechanical testing at relevant temperatures.

Project Description:

To move toward qualification of FeCrAl alloys for Accident Tolerant Fuel (ATF) cladding multiple tests need to be performed on the optimal FeCrAl alloys in tube form to obtain statistical data on mechanical properties at room temperature and elevated temperature. The optimal alloy being produced in tube form is C26M. In this research, specimens were cut directly from C26M tubing and tested in tension at room temperature, 300 and 600°C.

Accomplishments:

The axial tensile properties of C26M tube were successful measured at Los Alamos National Laboratory (LANL). Test were conducted at a constant strain rate of 10-3 s^{-1} and three different temperatures (22, 300, and 600°C). Tensile samples were machined from C26M tubes using wire electric discharge machining, with the nominal specimen geometry is shown in Figure 1. The sample geometry measurements and tensile testing were conducted following a similar procedure outlined in a previous

Figure 1. Nominal specimen geometry and machining location of axial tensile samples.

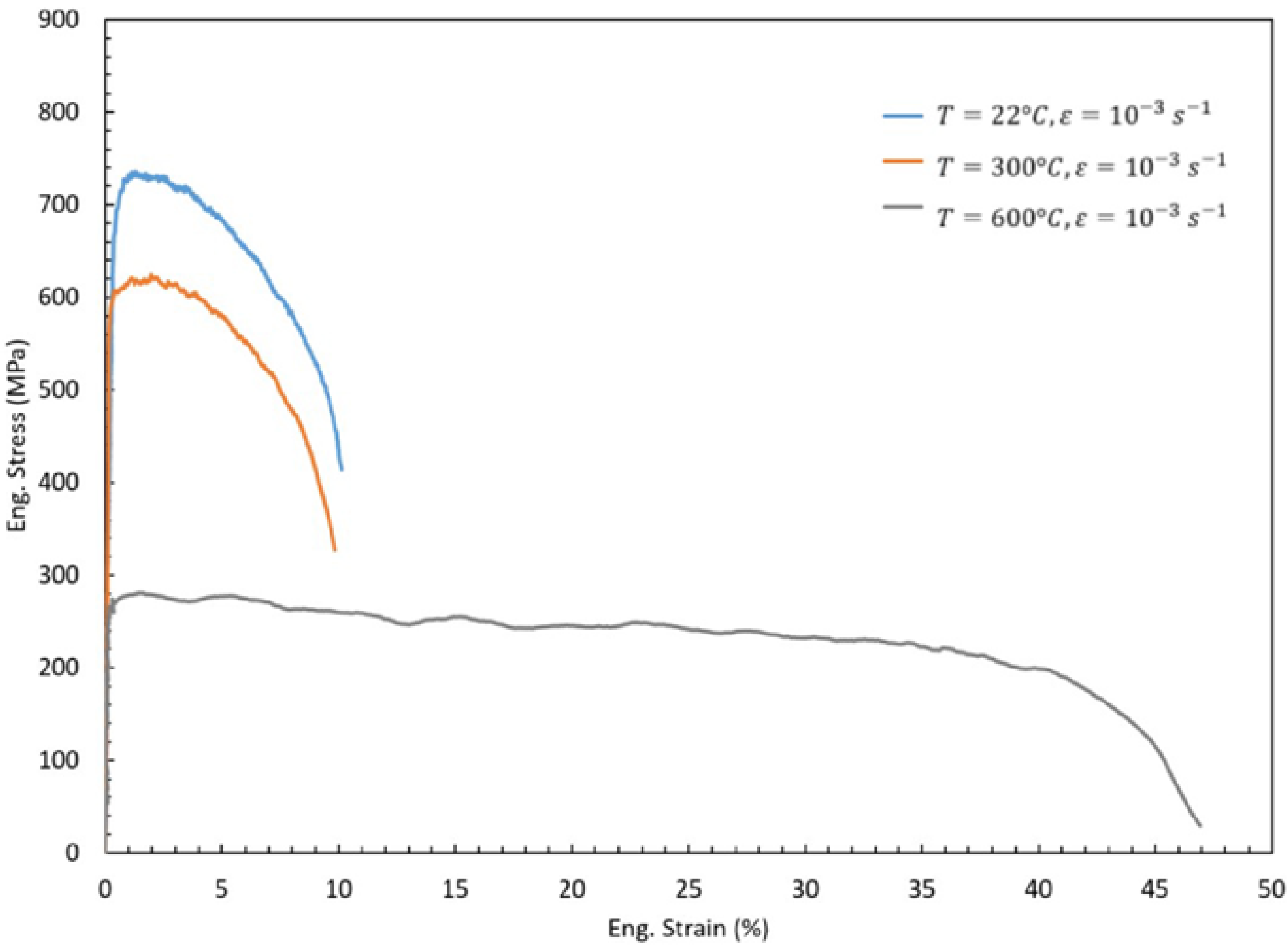


Figure 2. Engineering stress-strain curves for axial tensile specimens tested at 22, 300, and 600 degrees Celsius at a constant strain rate of 10^{-3} s^{-1}.

	YS (MPa)	YS (MPa)	UE (%)	TE (%)
22°C	710	730	1.2	10.3
300°C	605	637	1.6	9.4
600°C	273	290	1.4	47.1

Table 1. Average yield strength (YS), ultimate tensile strength (UTS), uniform elongation (UE), and total elongation (TE) for axial tensile specimens tested at 22, 300, and 600 degrees Celsius at a constant strain rate of 10^{-3} s^{-1}. Two samples were tested at each condition.

study [1]. A compliance correction was performed by applying a linear fit of the elastic section of the stress-strain curve and comparing this to a known published value [2] and then subtracting the calculated compliance from the experimental data.

Representative engineering stress strain curves of these test are shown in Figure 1. As typically expected, an increase in the testing temperature lead to a decrease in both the yield strength (YS) and ultimate tensile strength (UTS). The ultimate tensile strength was observed almost immediately post-yielding for all test conditions, and thus the uniform elongation is limited to 1.2-1.6%.

The total elongation (TE) of the 600°C specimens was notably higher (~47%) than both the 22 and 300°C specimens, which exhibited a similar response (~10%). Average values of the YS, UTS, uniform elongation (UE), and total elongation are shown in Table 1.

References:

[1.] T.J. Nizolek, B.P. Eftink, T.A. Saleh, Axial and Hoop Tensile Properties of FeCrAl C26M2 Tubing.

[2.] Z.T. Thompson, K.A. Terrani, Y. Yamamoto, Elastic Modulus Measurement of ORNL ATF FeCrAl Alloys, ORNL/TM-2015/632. (2015) 1–17.

Micromechanical testing of FeCrAl welds

Principal Investigator: Jonathan Gigax (MPA-CINT, LANL)
Team Members/Collaborators: Nan Li (MPA-CINT, LANL) and Stuart Maloy (MST-8, LANL)

The developed mesoscale mechanical testing technique and systems deliver bulk-like mechanical behavior at sizes orders of magnitude smaller, enabling localized testing of various mechanical properties including strength and ductility.

Because of the events related to the Fukushima reactor incident, innovative new cladding materials are being investigated for Light Water Reactors (LWRs) with improved resistance to accident conditions such as a Loss of Coolant Accident (LOCA). Materials need to be developed and tested to meet these challenging conditions. Some engineering alloys are presently available with promising properties but these alloys were not specifically developed for LWR applications. Thus, researchers at Los Alamos National Laboratory (LANL) are continuing development of a new improved accident tolerant fuel (ATF) cladding material (FeCrAl) in collaboration with Oak Ridge National Laboratory (ORNL) and performing micro and mesoscale mechanical testing on cladding and welds for this alloy.

Project Description:

ATF FeCrAl alloys are being developed at ORNL with optimal elemental constituents. Significant progress has been made in optimizing the alloy composition of Gen I and Gen II FeCrAl alloys and the optimum alloy is C26M. LANL is performing micro- and mesoscale mechanical testing on heats of C26M, including welds, to compare properties to previous alloys.

Accomplishments:

Projects and studies undertaken through the Advanced Fuels Campaign (AFC) have yielded a number of valuable advancements and insights, notably in small scale mechanical testing (SSMT). The need for larger tests that include more material volume and microstructural defects has been recognized, and motivated our foundational work to develop "mesoscale" mechanical testing platforms. These systems target specimen sizes on the order of 10's to 100's of micrometers. However, specimen preparation, especially at the larger end of the scale, proved challenging for conventional techniques, such as focused ion beam (FIB) and wire electro-discharge machining (EDM).

We (LANL and UC-Berkeley) developed techniques using femtosecond laser ablation to prepare a range of geometries commonly used in mechanical testing including pillars, tensile bars, and cantilevers (Figure 1) [1]. The principle advantage of this approach is the rapid removal of material that can access material thicknesses on the order of millimeters. However, the process is not damage free. Damage induced by the femtosecond laser ablation is minimal (less than 5 µm in pure Cu), making this a choice technique for coverage over a large range of sizes [2].

After establishing laser processing procedures, our attention focused towards investigating engineering materials subject to conventional

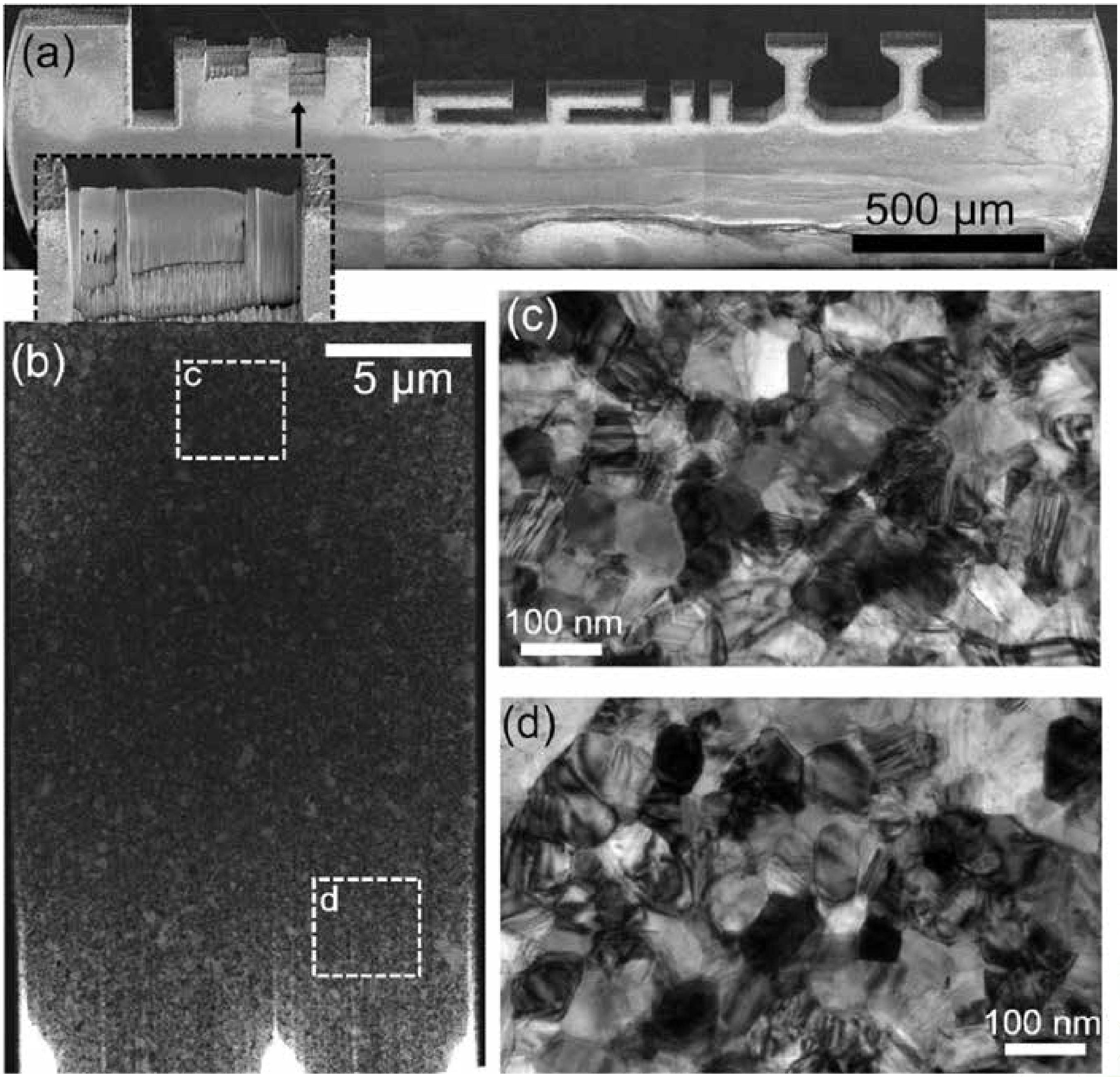


Figure 1. (a) Scanning Electron Microscopy (SEM) overview of a Transmission Electron Microscopy (TEM) grid femtosecond laser cut from a Physical Vapor Deposited (PVD) Cu foil. An inset of the thinned TEM region is marked by a dashed box. (b) Low magnification TEM image of one thinned region, with boxes labelled c and d corresponding to the respective high magnification images.

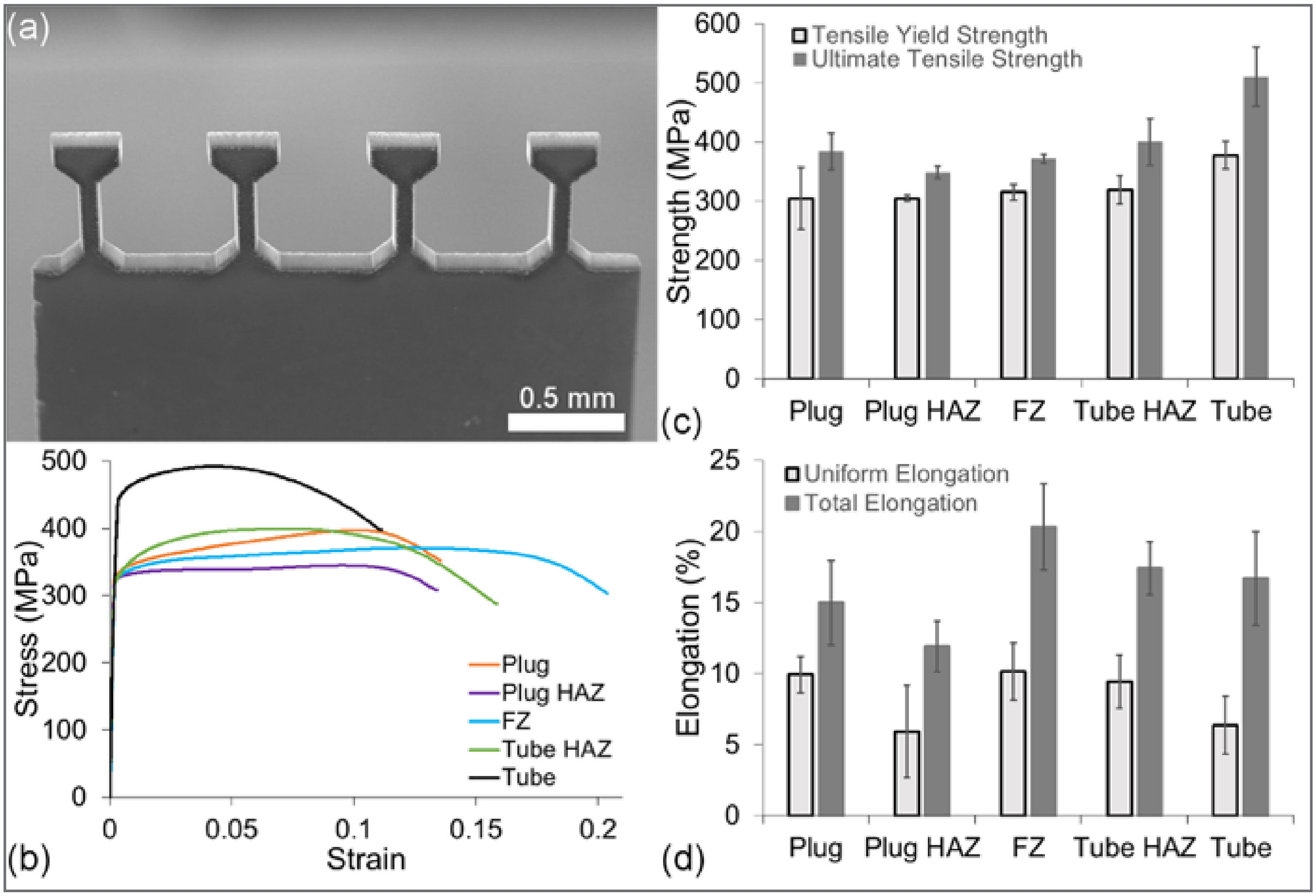


Figure 2. (a) SEM overview of the mesoscale tensile bars cut in the C26M weld specimen (45° tilt). Comparison of the (b) engineering stress-strain curves, (c) tensile yield and ultimate strength, and (d) uniform elongation and tensile ductility measured in each region.

tungsten inert gas (TIG) welding, in particular an accident tolerant fuel cladding alloy FeCrAl C26M. There have been a limited number of studies on FeCrAl welds, with the primary mode of mechanical testing involving straining the entire weld region and using digital image correlation to extract stress state information in the weld [3]. Leveraging our mesoscale test specimen sizes, we investigated the local properties of each weld region (i.e., heat-affected zones, fusion zone) [4].

Initial mesoscale tensile testing reasonably reproduced bulk-like responses, suggesting that conventional Taylor hardening models can be applied to understanding the deformation behavior from the microstructure at this length scale. The results of the study showed a number of interesting observations, as shown in Figure 2. While the strengths showed some variation in the weld regions, the uniform and total elongations differed significantly in the plug-side heat affected zone compared to the other regions. Coupled with the lowest measured strength, these results indicated that failure would occur first in this region.

Our team relies heavily upon nanomechanical testing, especially when investigating ion irradiated specimens, to study the changes to the mechanical properties. The smaller volume needed by these tests permits good sensitivity to the very shallow depths ions can penetrate. The weld study provided a good platform to benchmark the sensitivity of a variety of nanomechanical tests, including nanoindentation, micropillar compression, and microshear testing. It was found that nanoindentation with a Berkovich tip, best reproduced the trends observed from mesoscale tensile testing.

References:

[1] Q. McCulloch, J. Gigax, and P. Hosemann, JOM 72 (2020) 1694.

[2] J. G. Gigax, H. Vo, Q. McCulloch, M. Chancey, Y. Wang, S. A. Maloy, N. Li, and P. Hosemann, Scripta Mater. 170 (2019) 145.

[3] M. N. Gussev, K. G. Field, and Y. Yamamoto, Mater. Des. 129 (2017) 227.

[4] J. Gigax, A. Torrez, Q. McCulloch, H. Kim, N. Li, S. Maloy, J. Mater. Res. (2020), doi: 10.1557/jmr.2020.195.

2.4 LWR IRRADIATION TESTING AND PIE TECHNIQUES

Characterization of Irradiated Fuels with Pulsed Neutrons at LANSCE

Principal Investigator: Sven C. Vogel (LANL)
Team Members/ Collaborators: Alexander M. Long (LANL), D. Travis Carver (LANL), Jason Harp (ORNL), Luca Capriotti (INL), Eric J. Larson (LANL), James Angell (INL) and Aaron Craft (INL)

Characterization of irradiated fuels is typically done destructively to obtain samples with manageable radioactivity. Pulsed neutrons in combination with appropriate shielding casks allow characterization of entire irradiation capsules by diffraction, energy-resolved neutron imaging and neutron absorption resonance spectroscopy. In fiscal year (FY) 20 our team consisting of researchers from Los Alamos National Laboratory (LANL), Idaho National Laboratory (INL), and Oak Ridge National Laboratory (ORNL) made progress in technique development towards designing a shielding cask for neutron characterization as well as towards characterizing a 6mm diameter, ~2mm thick disk of an irradiated U-10Zr-1Pd fuel sample.

Project Description:

A key enabler for a "science-based approach" for accelerated development and certification of new, nuclear fuels, such as accident tolerant fuel forms, is an early and efficient understanding of material behavior under irradiation at multiple length scales. For new fuel formulations, there is a dearth of irradiation testing experience. When representative irradiation tests take years, it is important to extract the maximum insight possible from each test. Post irradiation examination (PIE) of fuels is mature and sophisticated but measurements in hot cells are expensive and typically examine small volumes of irradiated fuels relying on a limited amount of knowledge of the bulk characteristics of the specimen to select the volume examined destructively. This leaves the possibility that key failure inducing phenomena will not be observed because of limited PIE. For these reasons, there is value in techniques that can quickly and non-destructively characterize properties over volumes consistent with standard fuel geometries. Applying such methods pre- and post irradiation unambiguously determines the changes induced by irradiation. Such techniques can inform models on the initial condition of samples as well as provide the same parameters after the irradiation. Such a technique can complement, guide, and leverage destructive post irradiation examination by characterizing the entire sample. Pulsed neutron characterization, different from reactor-based neutron characterization, offers to fulfill this demand. For example, three dimensional characterization of complete fuel pellets within cladding after irradiation could identify regions that are representative of average and atypical response with parameters not accessible by presently applied techniques such as visual inspec-

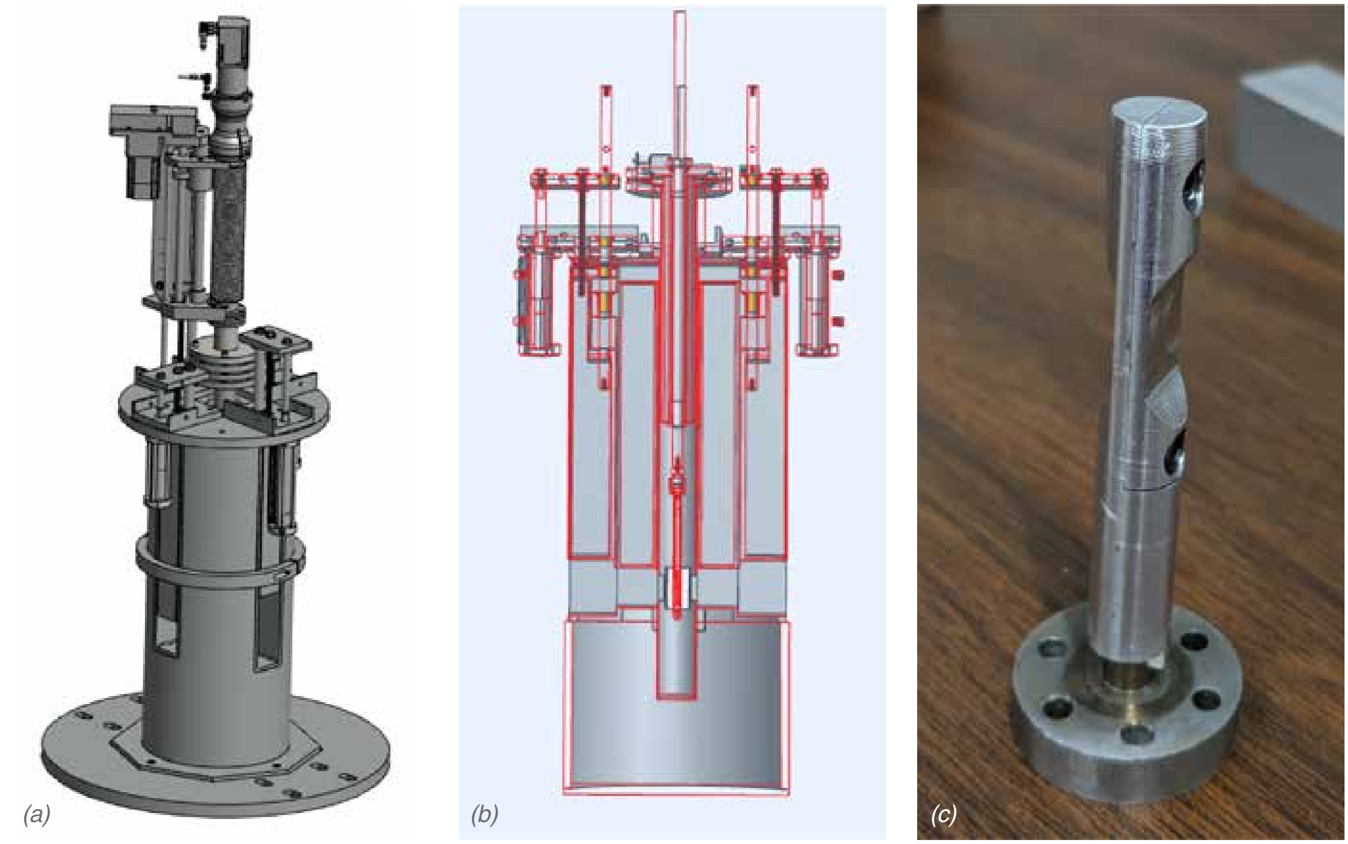


Figure 1. Schematic of the entire cask with motion control unit (a) and cross-sectional view to illustrate the beam path for incident and radiography neutron beams (b) (without motion control unit). The diameter of the cask is 15 inches and the lead layer in each direction of the sample is 10 cm. (c) Custom made aluminum sample holder for the irradiated 6mm diameter, 2 mm thick U-10Zr-1Pd sample.

tion, thermal neutron radiography, or acoustic methods. Ultimately, this development will provide pre- and post irradiation material characterization that is not presently available which will benchmark simulation codes and contribute to a solid scientific foundation for development and licensing of nuclear fuels. Building these capabilities is part of the activities of a work package to develop advanced post irradiation techniques.

Accomplishments:

To enable the aforementioned post irradiation characterization of bulk irradiated fuels, such as the AFC-2C/2D MOX fuels held back from destructive PIE for this purpose, shielding casks that allow to safely handle these fuels are required. Dose rates of ~900R/hr are anticipated for these particular samples, imposing substantial demands on the cask design. Therefore, the design of such a shielding cask is driven by a variety of criteria ranging from reducing the dose rates for the staff handling such a cask to acceptable levels to allowing neutron characterization, sample mounting in hot cells, prevent spread of contamination, fitting of the cask into a Department of Transportation (DOT)-approved transport cask for shipment between INL and LANL, remaining intact in various accident scenarios, the ability to cask lead into the steel sheel, and having a weight suitable for the involved rigging equipment. These criteria and the resulting design are described in detail in a milestone report entitled "Status Report on Development of a Cask to Enable Pulsed Neutron Characterization of Irradiated Fuel", M4FT-20LA020201022. While the core of the development was done by the team at LANL, design reviews with collaborators from INL and ORNL were conducted. Besides the AFC-2C/2D MOX fuels, the cask is capable to handle other fuel geometries with a maximum diameter of 0.5 inches and maximum capsule length of 6 inches.

Figure 1 (a) shows a schematic of the cask sitting on a base plate attaching to alignment equipment. The pneumatically actuated shutter windows for incident and diffracted beam are open in the schematic. On top of the cask is the motion stage, allowing for accurate vertical motion to select a volume along the sample length and rotation for tomography and texture measurements. The sectional view in Figure 1 (b) shows the volume in which the sample resides as well as the beam paths provided by the opened shutters. A prototype of the cask without the lead is fabricated and scheduled to be commissioned in FY21 in beamlines. Dry-runs of sample mounting in INL hot cells are also planned. Characterization of irradiated fuels with this cask is planned as early as FY22. Support for the design and prototype fabrication through a FY20 LANL Technology Evaluation and Demonstration grant is gratefully acknowledged.

As an independent step to establish procedures at LANL to characterize irradiated and therefore highly radioactive samples with pulsed neutrons at the Los Alamos Neutron Science Center (LANSCE), procedures were developed to characterize an irradiated U-10Zr-1Pd sample irradiated in the Advanced Test Reactor (ATR). This sample has a dose rate of ~3R/hr on contact and can be handled with

The advanced post irradiation examination package moves towards characterizing entire irradiation capsules with pulsed neutrons to provide more data for modelers and to help pick the most interesting volumes for destructive PIE.

remote handling. It has a diameter of 6mm and 2mm thickness and is enclosed in a custom Al sample holder. Measurement in the high pressure/preferred orientation (HIPPO) neutron time-of-flight diffractometer as well as the energy-resolved neutron imaging (ERNI) beamline were demonstrated. The energy-resolved neutron imaging allows to set contrast to specific isotopes using the neutron absorption resonance phenomenon, very strong absorption of neutrons of a specific neutron energy. The pulsed neutron spallation source at the LANSCE in combination with a pixiliated time-of-flight imaging detector allows to measure the neutron energy by the time-of-flight principle, a unique advantage over constant-wave reactor-based neutron radiography. A 1 mm^2 collimator was built and installed to limit the field of view on the disk-shaped sample. With this collimation and a liquid scintillator detector installed at the 60 m flightpath length station of this beamline, scans across the sample will be enabled that provide a much larger energy range for the resonance analysis at the cost of a reduced spatial resolution. A first application of this newly developed method to a Nuclear Science User Facilities (NSUF)-funded full characterization of the U-10Zr-1Pd sample is planned for FY21 as a collaboration between LANL, INL and ORNL.

Completion of ATF-1 Experiment Series

Principal Investigator: David Kamerman
Team Members/ Collaborators: Chris Murdock, Klint Anderson, Jason Brookman and Changhu Xing

The ATF-1 experiment allowed for the prioritization of leading ATF concepts such as coated Zircaloy and FeCrAl cladding.

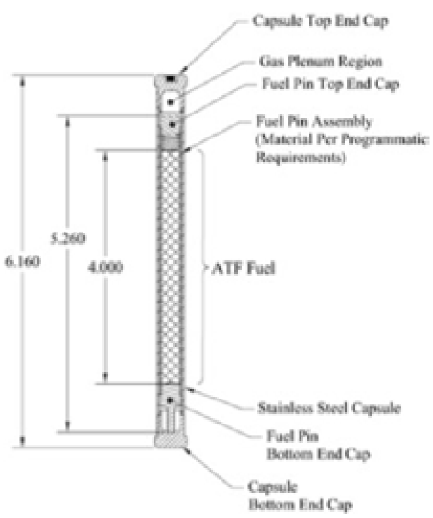


Figure 1. ATF-1 Rodlet and Capsule Design

The ATF-1 experiment was part of the "Feasibility and Assessment" phase of the Accident Tolerant Fuel (ATF) program. Test rodlets made of entirely new fuel and cladding materials were tested in Drop-in-Capsule experiments in the small I test postions in the Advanced Test Reactor (ATR) reflector. The test design provide prototypic nuclear and thermal conditions but without the hydraulic environment found in light water reactors.

Project Description:

The ATF-1 experiment test train consists of the basket, top spacer, and seven vertically-stacked capsule assemblies and/or dummies in each of the three channels (for a total of twenty one (21) capsules/dummies per basket). The capsule used in the experiment provides the pressure boundary for the experimental materials located within. In the event of a rodlet cladding failure, the stainless steel (316L) capsule prevents fission products from entering the ATR Primary Coolant System (PCS). The test rodlet is intended to represent a miniature length Pressurized Water Reactor (PWR) fuel rod, nominally prototypic in the radial dimension consisting of 10 pellets. A typical ATF 1 capsule and rodlet assembly is shown in ATF-1 Rodlet and Capsule Design below. Concepts irradiated included Uranium Silicide and Uranium Nitride fuels, Uranium dioxide with Silicon Carbide dopants as a fuel, and FeCrAl cladding. Irradiations began in January of 2015 in ATR cycle 157C and concluded in January of 2020 after ATR cycle 166B. The rodlets were irradiated at prototypic linear heating rates between 200 – 400 W/cm. Cladding temperature was controlled using a gas gap between the rodlet and capsule.

Accomplishments:

A total of 31 test rodlets were irradiated in the ATF-1 experiment with the highest burnup pin being a Uranium Silicide pin that reached 45 GWd/MTU. Testing showed that Uranium Silicide performed well under irradiation with minimal swelling and fission gas release. However, this fuel concept has been discontinued due to the materials unfavorable reaction with water. Testing on the SiC doped UO_2 was less favorable with high amounts of porosity seen in the pellets even at low burnups. Burnups and power histories for all of the rods involved in the experiment are available and have been provided to the respective industry parties. The power history of the ATF-07 capsule which contained a FeCrAl clad fuel rod is shown in ATF-07 Power History. Cladding temperature data can be determined from a polynomial correlation with the rod power. The correlation is derived from a finite element model of the rodlet and capsule. An example relationship for the ATF-07 capsule is shown in Cladding Temperature Correlation for ATF-07. Further test data can be found on the ATF database website at atf.inl.gov. The design of

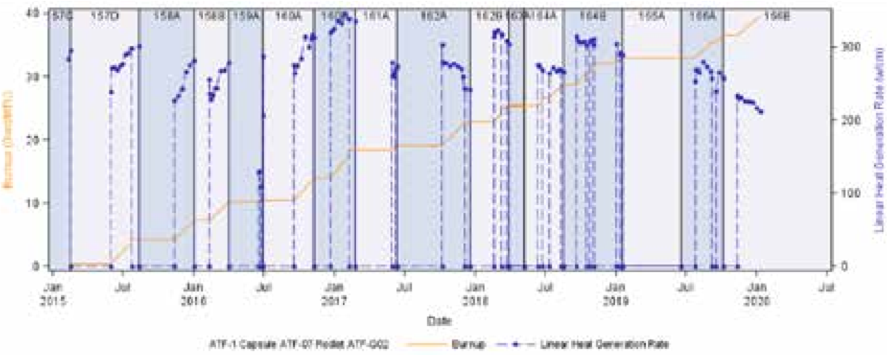


Figure 2. ATF-07 Power History

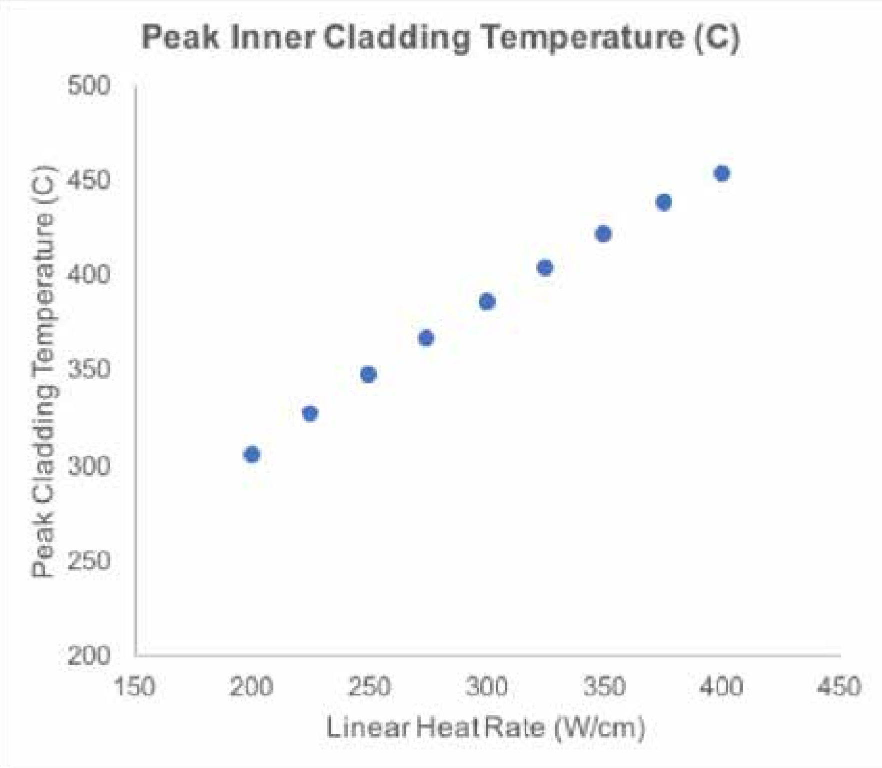


Figure 3. Cladding Temperature Correlation for ATF-07

the ATF-1 experiment is being used for future irradiation experiments that will be used to develop specific licensing data for Chrome doped UO_2 fuel and SiC-SiC cladding. These future tests are set to begin in January of 2021 in ATR cycle 169A.

Status of ATF-2

Principal Investigator: David Kamerman
Team Members/ Collaborators: Gary Hoggard, Brian Durtschi, Bryon Curnutt and Stacey Wilson

Irradiation of prototypic rodlets at conditions that are representative of commercial reactor environments is critical to the development of fuel performance data that can be used to develop licensing topical reports.

The ATF-2 experiment is part of the "Development and Qualification" phase of the Accident Tolerant Fuel (ATF) program. Test rodlets of mature ATF technologies are irradiated in a prototypic pressurized water loop in the center flux trap of Advanced Test Reactor (ATR) at Idaho National Laboratory (INL).

Project Description:

The ATF-2 experiment consists of 6 tiers of test rod holders each containing a 2x3 grid of 15 cm test rodlets. A graphical rendering of the test train is shown in Figure 1. Each of the test rodlets contain 10 pellets of fuel each of which is approximately 1cm long. Tiers 5 and 6 are combined into a single tier of longer test rods at the top of the test train. The test rodlets are supplied by each of the three vendor teams and consist of their leading and most mature ATF concepts. Concepts irradiated to date include coated cladding, iron chrome aluminum cladding, and doped fuel pellets. Irradiation began in June of 2018. The first three cycles (164A, 164B, and 166A) ran at lower linear heat generation (LHGRs) (180-240 W/cm). Subsequent cycles have been running at high power (350-400 W/cm). The first two cycles (164A and 164B) ran with lower flow rates of approximately 35gpm and cooler inlet temperatures of approximately 250°C. Subsequent cycles ran with higher flow rates of approximately 50gpm and inlet temperatures of approximately 280°C. Loop thermal hydraulic data is available for each cycle. An example, cycle 168A, which had numerous mid-cycle shutdowns is shown below in Figure 2.

Accomplishments:

Forty-one rodlets have been irradiated to date and the highest burnup rod is projected to reach over 30 GWd/MTU by the end of cycle (EOC) 169A prior to ATR's core internal changeout (CIC). Power histories are evaluated at the end of each cycle following

Table 1. Burnup Accumulation of All Rods Irradiated in ATF-2 Before ATR Core Internal Changeout"

	164A (6/12/18-8/17/18)	164B (9/19/18-1/17/19)	166A (7/25/19-9/24/19)	166B (11/11/19-1/10/20)	168A (4/15/20-7/22/20)	168B (8/20/20-10/19/20)	169A (2/5/21-4/6/21)
R-04	2.6	5.8	9.3				
R-05	2.6	5.8	9.3				
R-06	2.7	6	9.5				
R-07	2.6	5.8	9.2				
R-08	2.7	5.9	9.4				
R-09	2.6	5.9	9.4				
R-10	3	6.5	10.3				
R-11	2.9	6.4	10	14.9	19.63	23.9	28
W-01	2.3	5.2	8.1				
W-02	2.3	5.1	8				
W-03	2.4	5.2	8.2				
W2-02					4.8	9.2	13.5
W2-03					4.8	9.1	13.3
W2-05					4.7	8.9	13

INL's engineering process to produce high quality data. Power histories have currently been evaluated through cycle 166B and a plot of test rod AR-15's power history is shown below in Figure 3. Further test data can be found on the ATF database website at atf.inl.gov. A list of all the test rods irradiated to date and their burnup accumulation is shown below in Table 1 along with preliminary results from cycle 168A and projects for the next two cycles before CIC 168B and 169A.

	164A (6/12/18-8/17/18)	164B (9/19/18-1/17/19)	166A (7/25/19-9/24/19)	166B (11/11/19-1/10/20)	168A (4/15/20-7/22/20)	168B (8/20/20-10/19/20)	169A (2/5/21-4/6/21)
GE-IC-04				4.7	9.4	13.9	18.2
GE-IC-05				4.6	9.2	13.5	17.7
GE-IC-07				4.8	9.5	14	18.4
GE-IC-08				18.2	17.7	18.4	18.9
GE-IC-09				4.8	9.6	14	18.4
GE-IC-10				5	9.9	14.5	19
GE-ZR-18				4.8	9.7	14.1	18.6
GE-ZR-20				5	10	14.7	19.2
GE-A0-07				4.7	9.5	13.9	18.2
GE-A1-07				4.9	9.8	14.4	18.9
GE-A0-08				5	9.9	14.5	19
GE-A1-08				5.1	10.3	15	19.7

	164A (6/12/18-8/17/18)	164B (9/19/18-1/17/19)	166A (7/25/19-9/24/19)	166B (11/11/19-1/10/20)	168A (4/15/20-7/22/20)	168B (8/20/20-10/19/20)	169A (2/5/21-4/6/21)
AR-01	2.6	5.8	9.1	14.8			
AR-02	2.6	5.8	9.2	14.9			
AR-03	2.9	6.3	9.9	14.4	19.8	24.7	29.3
AR-04	2.3	4.9	7.8	12.5	17.9	22.7	27.3
AR-05					5	9.6	14
AR-06					4.9	9.3	13.6
AR-11	2.6	5.6	9	14.4			
AR-12	2.6	5.8	9.3	15	20.5	25.4	30.1
AR-13	2.6	5.8	9.2	14.8	20.2	25	29.7
AR-14	2.6	5.7	9	14.3			
AR-15	3	6.5	10.2	15	19.7	24	28
AR-16	3	6.5	10.2	14.9	20	24.5	28.9
AR-17	2.3	5	7.8	12.4	17.5	22.1	26.5
AR-18					4.9	9.3	13.6
AR-19					5.1	9.7	14.2

Figure 1. Graphical Rendering of ATF-2 Test Train

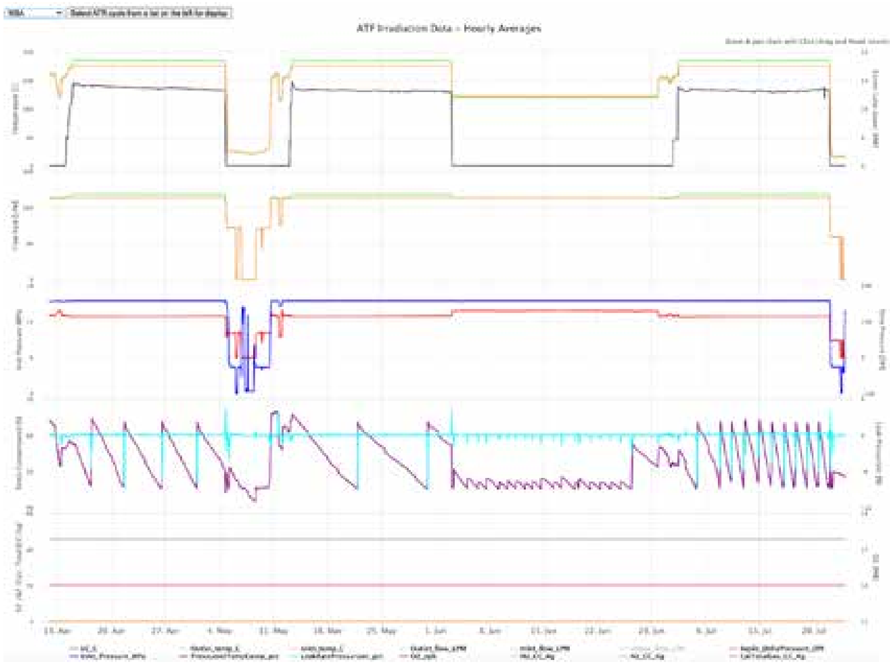


Figure 2. Loop 2A Thermal Hydraulic Data for Cycle 168A

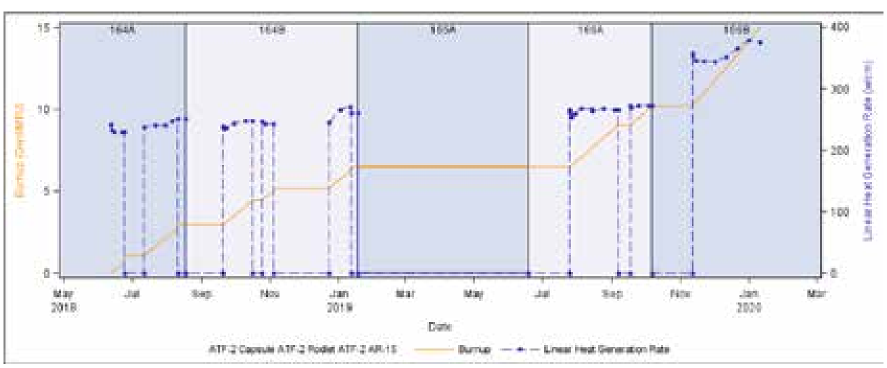


Figure 3. Power History of AR-15

MiniFuel Irradiation of Monolithic Fuels

Principal Investigator: Christian Petrie
Team Members/ Collaborators: Annabelle Le Coq, Dylan Richardson, Chris Hobbs, Grant Helmreich, Joseph Burns and Jason Harp

This is the first irradiation of monolithic ceramic fuels using the MiniFuel vehicle, which offers high throughput, economical, separate effects irradiation testing capabilities with rapid burnup accumulation to support accelerated fuel qualification.

Qualification and deployment of any new fuel requires rigorous irradiation testing to demonstrate performance under representative normal and off-normal operating conditions. The traditional approach for qualifying new fuels requires exhaustive execution of many integral fuel tests. However, due to the long timeframe for executing these integral tests and the limited number of available materials test reactors, this approach is becoming impractical. To accelerate the timeframe to qualify new nuclear fuels, a new approach is being developed that relies on modern modeling and simulation tools to rapidly identify parameters with a high impact on fuel performance and a large uncertainty. This data will allow for proper prioritization of targeted, separate effects irradiation experiments. To this end, Oak Ridge National Laboratory (ORNL) developed the MiniFuel irradiation vehicle for use in conducting accelerated separate effects irradiation testing of a wide range of nuclear fuel materials in the High Flux Isotope Reactor (HFIR) [1]

Project Description:

Separate effects irradiation tests are designed specifically to isolate the most impactful fuel performance variables and provide experimental data to fill these critical gaps in the fuel performance models. The MiniFuel irradiation vehicle allows for highly accelerated burnup accumulation with minimal coupling between the fission rate and the fuel temperature. This is accomplished by reducing the volume of the fuel and packaging the miniature fuel specimens inside individually sealed capsules. By ensuring that the total nuclear heat generated inside each capsule is dominated by gamma heating in the structural components—as opposed to fission heating in the fuel—this allows for a flexible irradiation vehicle that can accommodate a wide range of fuel compositions, enrichments, and even geometries. The small size of the fuel also greatly reduces temperature gradients, which allows for near-isothermal conditions. The first MiniFuel irradiations that were performed tested sol gel–derived uranium nitride kernels and tristructural isotropic (TRISO)-coated particle fuels. While particle fuels are being considered as an advanced fuel form for both light water reactors (LWRs) and various advanced reactor concepts, the ability to test monolithic fuels such as doped and undoped UO_2, as well as other ceramic fuels, is critical to evaluating more near term accident-tolerant fuel (ATF) concepts and to support burnup extension. This work represents the preparation and assembly of the first set of monolithic MiniFuel irradiations, which included U_3Si_2 and reference UO_2 disk specimens [2].

Accomplishments:

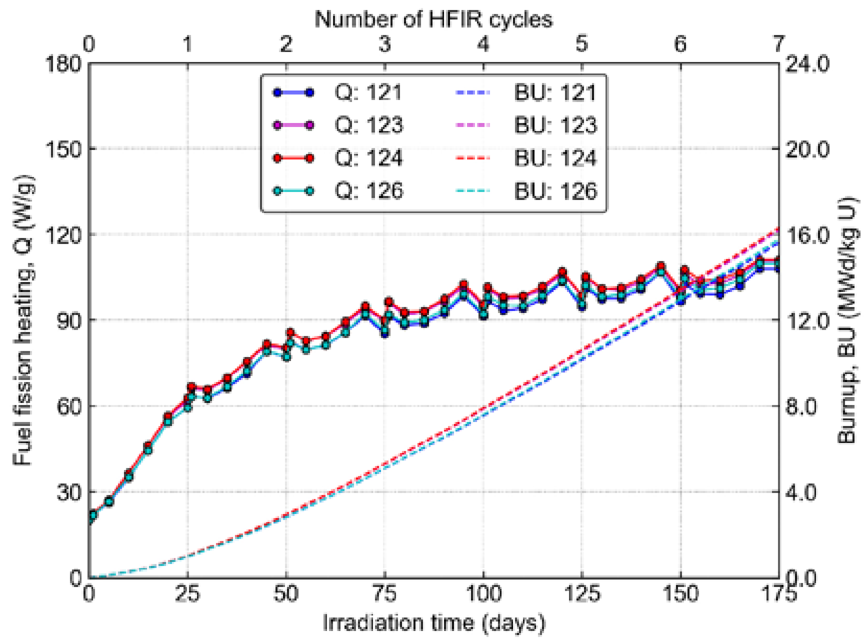
Two irradiation targets containing a variety of UO_2 and U_3Si_2 disk fuel specimens were fabricated and assembled for irradiation to targeted burnups of 8–10 and 28–40 MWd/kg U. The target irradiation temperature is 450–550°C. The UO_2 fuel disks were

Figure 1. Capsule assembly pictures showing a fuel specimen placed in a cup (top left) and loaded into a capsule with a TM (top center, right). Six capsules are stacked together (middle) and welded inside a steel target (bottom).

fabricated at ORNL using a traditional press and sinter approach with densities generally in the range of 92–97% as determined using geometric measurements, x-ray computed tomography (XCT), and a 3D optical profilometer [2]. The U_3Si_2 disks were fabricated at Los Alamos National Laboratory (LANL) using standard powder metallography techniques with densities generally in the range of 90–97% [3] as determined using the same techniques. If XCT and/or optical profilometry can be used post irradiation, this could allow for a direct volume comparison to quantify radiation-induced swelling.

The monolithic MiniFuel capsules were successfully assembled, welded, and tested per HFIR requirements and one of the two targets was inserted into HFIR starting in cycle 487 (April 2020). Figure 1 shows pictures of the experiment assembly process. The fuel is loaded inside a molybdenum cup (top left) and placed inside of a molybdenum filler. This assembly is loaded inside the capsule (top right), which is welded in helium after inserting a passive SiC temperature monitor, or TM (top middle). Six capsules are assembled in each stainless-steel target (middle), which

Figure 2. Calculated evolution of fission rate and burnup over the first 7 cycles of irradiation for the U3Si2 disk specimens for different RAC locations indicated in the legend.



is welded in a helium-argon mixture (bottom). Figure 2 shows the calculated burnup and fission rate evolution for the U_3Si_2 specimens over the first 7 cycles of irradiation. The numbers (RAC) indicated in the legend indicate the radial position (R=1, facing away from the HFIR core), the axial target position (A=2, core midplane), and the capsule position (C) within the target, numbered 1–6 from bottom to top. The fission rates appear to approach a steady-state value in the range of 100 W/g after reaching an equilibrium between breeding and burning of Pu isotopes. The first (higher burnup) target is expected to finish irradiation in early calendar year 2022. The second (lower burnup) target has not yet been inserted due to limited position availability in HFIR. This target is expected to start irradiation in early calendar year 2021 and finish irradiation during the fall of the same year. Figure 3 shows temperature contour plots (in °C) for a U_3Si_2 specimen and its surrounding capsule components. Temperature variations within the fuel specimens are calculated to be <5°C. Figure 4 shows calculated average fuel and TM temperatures at beginning of cycle (BOC) and end of cycle (EOC) for cycles 1, 4, and 12. As expected, the average fuel temperatures are constant to within <40°C by cycle 4 once equilibrium fission rates have been achieved. The calculated TM temperatures are <30°C lower than the fuel temperatures. Irradiation of U_3Si_2 will provide new data regarding the irradiation performance of a candidate ATF to complement current ATF-1 integral

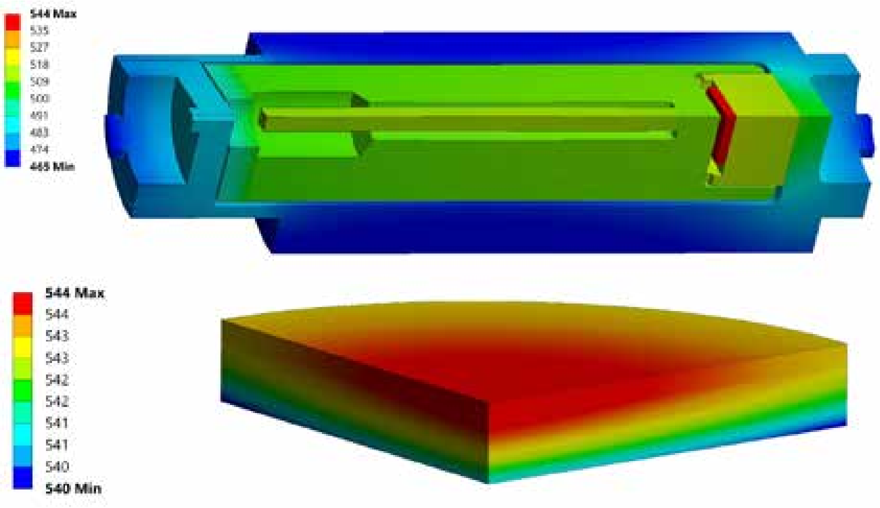


Figure 3. Calculated temperature contours (°C) for one capsule (top) and its U_3Si_2 disk specimen (bottom).

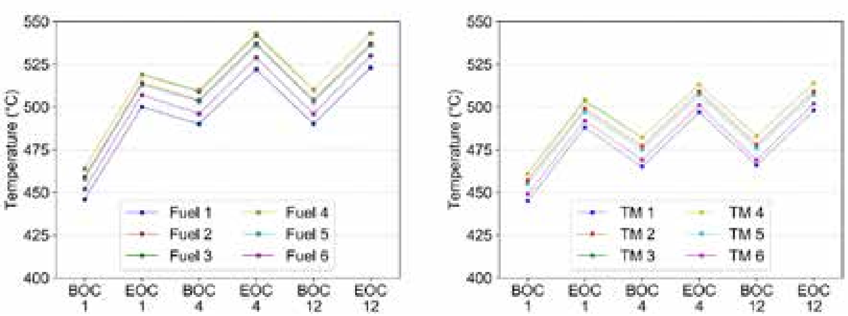


Figure 4. Calculated evolution of the average fuel (left) and TM (right) temperatures at the beginning of cycle (BOC) and end of cycle (EOC) for HFIR cycles 1, 4, and 12.

experiments being performed in the Advanced Test Reactor. UO_2 samples were included as a reference so that the results from the MiniFuel experiments can be compared with the extensive UO_2 fuel performance database.

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PIE of Coated Cladding rods from ATF-2

Principal Investigator: Fabiola Cappia (INL)
Collaborators: Kiran Nimishakavi (Framatome), Russ Fawcett (Global Nuclear Fuel) and Luke Olson (Westinghouse)

Coated zirconium alloy fuel cladding concepts are being pursued by several U.S. fuel vendors as part of the U.S. Department of Energy's Accident Tolerant Fuel (ATF) program. Coated zirconium alloys are near-term evolutionary concepts with potential of allowing longer coping times during accident scenarios while maintaining or improving cladding performance during normal operation and operational transients.

Project Description:

Better performance of the cladding and mitigation of detrimental phenomena such as oxidation and hydrogen pick up can result in both increased safety margin and longer exposure times without operational penalty. For instance, out-of-pile testing of coated cladding have shown that the coating provides enhanced protection of the fuel rod against wear and fretting and improved oxidation resistance, both during normal operation and during accidents [1–3]. While tests on non-irradiated material are fundamental for value assessment of the ATF candidates, in-pile tests under prototypical Light Water Reactor (LWR) conditions such those of the ATF-2 loop and subsequent post irradiation examination (PIE) are needed to confirm and complete the former. The goal of the PIE on the coated rods is to verify the performance of these ATF concepts with focus being on corrosion behavior, irradiation-induced microstructural evolution and mechanical properties of the cladding. The data gathered are of fundamental importance to validate material models, to confirm the improved performance anticipated by out-of-pile testing and to build the PIE database that can support qualification and licensing of the new concepts.

Each of the industry partner is pursuing a coated cladding ATF candidate. Westinghouse and Framatome are focusing on Cr-coating of their commercial pressurized water reactor (PWR) cladding concepts, specifically M5® and Zirlo®, while Global Nuclear Fuel (GNF) is pursuing ARMOR™-coated Zircaloy-2 cladding. The rodlets were inserted for irradiation in the Advanced Test Reactor (ATR) pressurized loop starting in 2018. The first rodlets completed irradiation at the end of 2019 and PIE have been conducted on these first rodlets throughout the fiscal year.

Accomplishments:

The PIE took place in the Hot Fuel Examination Facility (HFEF) at Idaho

The PIE results of this irradiation campaign provide critical data for each vendor team to assess the performance of coated cladding and to receive licensing approval for their ATF near-term concepts.

National Laboratory (INL). The PIE campaign started with non-destructive examinations, which included rodlet visual inspection, neutron radiography, gamma scanning, and profilometry measurements. All the non-destructive examinations have been completed on the rodlets received during the year. The PIE campaign will continue with the destructive and advanced examinations in the coming years.

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2.5 LWR FUEL SAFETY TESTING

Development and Demonstration of a Methodology to Evaluate High Burnup Fuel Susceptibility to Pulverization under a Loss of Coolant Transient

Principal Investigator: Nathan Capps (Oak Ridge National Laboratory)
Team Members/ Collaborators: Ryan Sweet (Oak Ridge National Laboratory; University of Tennessee, Knoxville), Brian Wirth (University of Tennessee, Knoxville), Andrew Nelson (Oak Ridge National Laboratory) and Kurt Terrani (Oak Ridge National Laboratory)

This methodology is a first of a kind using high fidelity Department of Energy (DOE) codes to investigate Light Water Reactor (LWR) High Burnup-related phenomenon.

For economic reasons, the US nuclear industry is renewing efforts to build a technical basis to extend rod average burnup limits above the current regulatory burnup limit of 62 GWd/MTU. The primary driver is to increase pressurized water reactor cycle lengths to 24 months, reducing the number of fresh fuel assemblies and core design constraints, thereby making core energy utilization more efficient. However, fuel pellet fragmentation and pulverization, termed high burnup fuel fragmentation (HBFF), has been observed in the high burnup (>90 GWd/MTU) Halden loss-of-coolant-accident (LOCA) integral test series. The issue gained attention when fuel fragmentation and pulverization were also observed closer to the current US regulatory limit during the US Nuclear Regulatory Commission (NRC) sponsored out-of-core integral test at Studsvik Nuclear in early 2011. This led to NRC concerns with potential changes to fuel and core designs relative to fuel pellet pulverization. In a letter to the NRC Commissioners, the staff specifically identified a need to "…define the boundary of safe operation for key fuel design and operating parameters," stating that "the staff is challenged to evaluate the acceptability of future fuel design advancements and fuel utilization changes." As such, it can be concluded that HBFF and potential dispersal into the reactor coolant system introduces additional complications in LWR fuel safety evaluations.

Project Description:

It is not clear how much fuel will be susceptible to HBFF; nor has there been a methodology developed to evaluate fuel susceptibility to HBFF. To that end, the paper proposes an analysis methodology to assess fuel susceptibility to HBFF during LOCA scenarios. The work presented used the BISON fuel performance code to evaluate a representative pressurized water reactor fuel rod exposed to a rod average burnup of 75 GWd/MTU. Sensitivity studies investigated the impact of the peak cladding tempera-

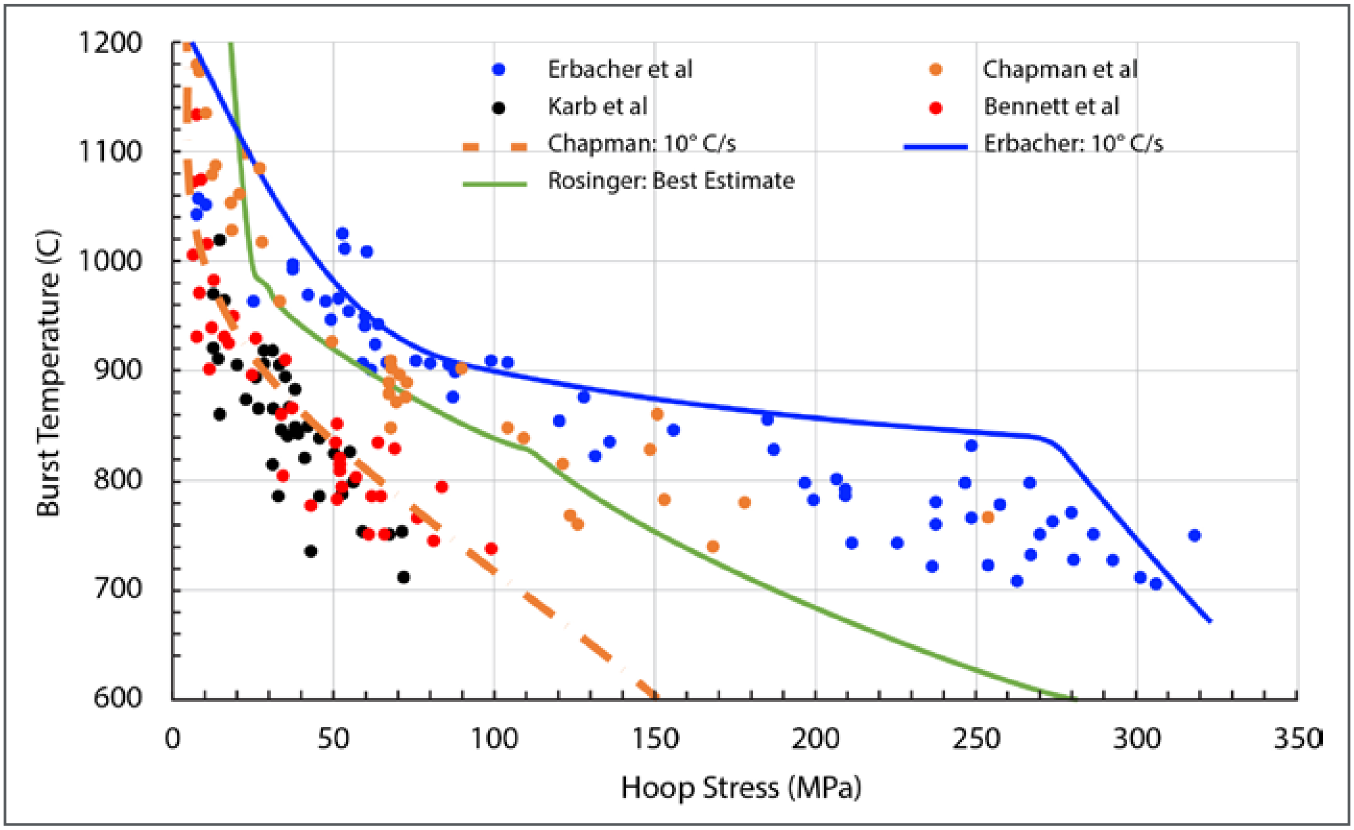

Figure 1. Comparison of cladding burst temperature as a function of hoop stress using several different data sources and models.

ture, transient fission gas released, and pre-transient fission gas release on cladding ballooning and burst timing. Subsequently, a methodology to assess fuel susceptibility to HBFF was developed based on experimental data published in the open literature. The methodology was demonstrated by calculating the mass of fuel suscepti-bility to HBFF.

Accomplishments:

This manuscript describes the development and demonstration of a methodology to assess HBFF susceptibility under LOCA condi-tions. The BISON fuel performance code was used to perform a generic pressurized water reactor (PWR) fuel rod analyses for a rod average burnup

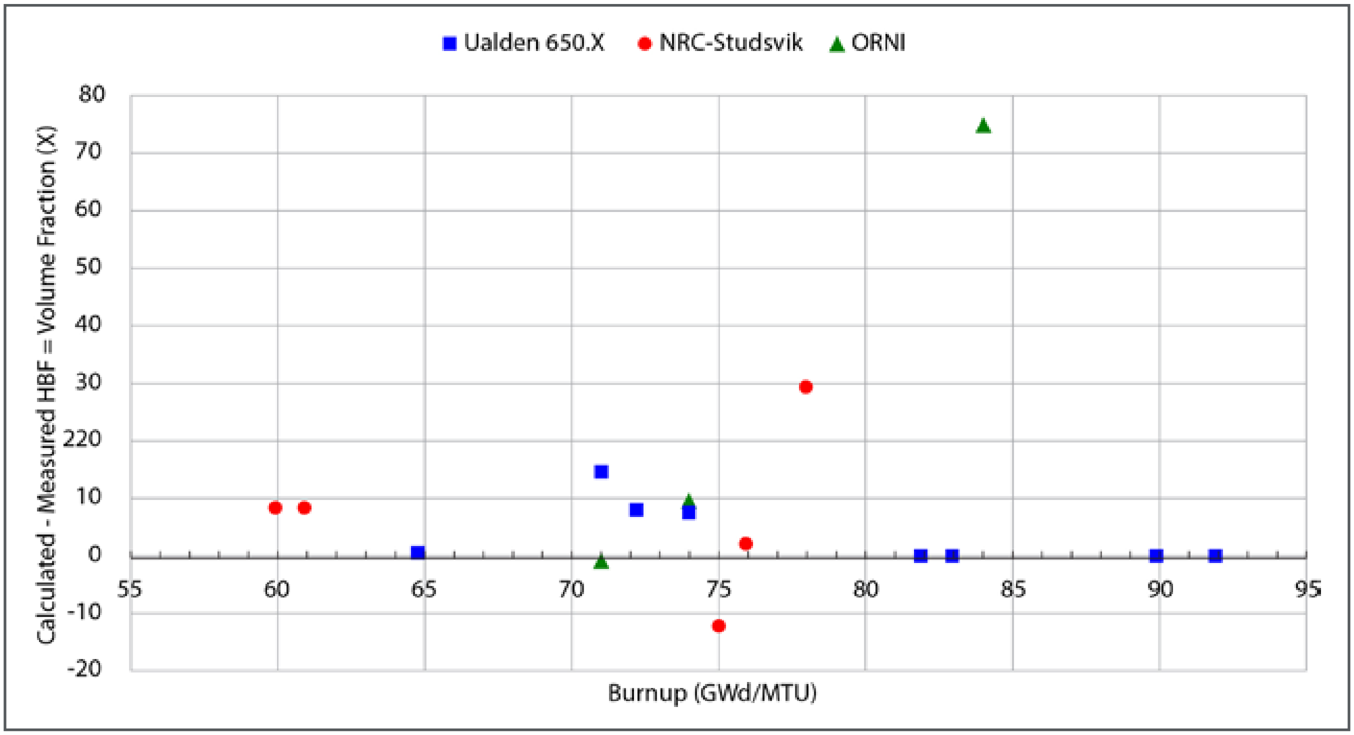

Figure 2. Pulverization threshold developed by Turnbull et al. validated to all publicly available integral LOCA test performed on high burnup fuel.

of 75 GWd/MTU. Several sensitivity studies were performed to gain a deeper understanding of the mechanisms that could lead to or reduce the amount of fuel susceptible to HBFF. The first study evaluated the impact of peak cladding temperature (PCT). The results indicate that (1) increasing PCT drastically decreased time to failure, and (2) calculated balloon size was consistent with the literature. Secondly, the effect of pre-transient and transient rod internal pressure on cladding balloon and burst was evaluated. For both scenarios, time to failure decreased as rod internal pressure increased, and the size of the balloon was relatively consistent. It was noted that large changes in time to failure of ~10 seconds resulted in a significant decrease in balloon size. Lastly, fuel susceptibility to HBFF was calculated for two cases: 800°C and 1,000°C. The lower temperature case resulted in significantly less fuel being susceptible to HBFF, and both

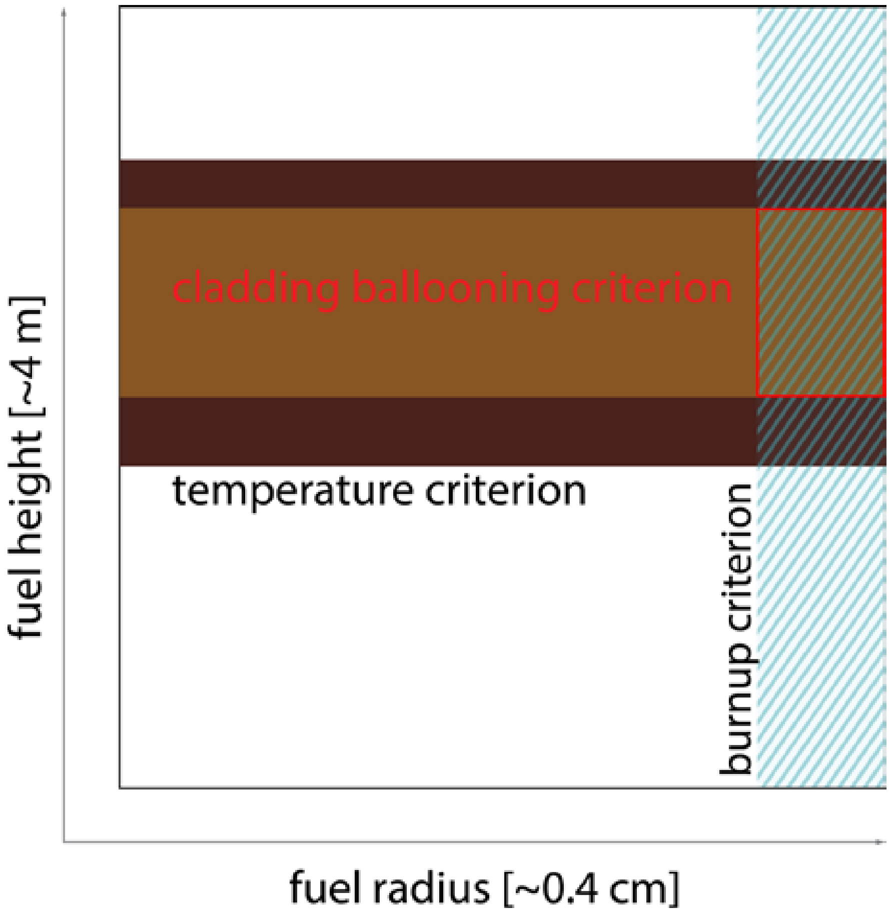

Figure 3. The mass of the fuel susceptible to HBFF is within the region that is enclosed in the red rectangle.

cases were determined to be overly conservative. Future multiphysics analyses and additional prototypic fuel fragmentation data were identified to further accelerate the industry's safety case. The methodology is currently being deployed through the Nuclear Energy Advanced Modeling and Simulation (NEAMS) program in collaboration with Southern Nuclear Company. Other industry partners have subsequently reached out to deploy the methodology using their licensed tools.

Boiling Transition (Dryout) Testing of FeCrAl Cladding

Principal Investigator: Ken Kane
Team Members/ Collaborators: Soon Lee, Sam Bell, Bruce Pint and Nick Brown

A novel experiment has been designed to simulate cyclic dryout has proven to be effective and pertinent, easily demonstrating the mechanical superiority of C26M over Zircaloy-2.

A novel experiment designed to simulate cyclic dryout in boiling water reactors has been developed utilizing the severe accident test station (SATS).

Project Description:

Cyclic dryout is simulated by modifying the out-of-cell Loss of Coolant Accident (LOCA) burst test. Testing was performed to determine the temperature cycling capabilities of the LOCA burst rig. After capabilities were determined, internally pressurized claddings were subjected to rapid thermal cycling in a steam environment until cladding fails via burst. Lifetime during cyclic dryout was found to be correlated with upper thermal cycle temperature, internal pressure, and cladding material.

Accomplishments:

Testing was performed on C26M FeCrAl and Zircaloy-2 cladding to determine lifetime during cyclic dryout as a function of thermal cycle and internal pressure. It was first necessary to determine the fastest possible heating rate using the infrared (IR) furnace. The maximum heating was determined to be 15-17°C/s, which is significantly higher than the 5°C/s heating rate used during standard out-of-cell LOCA burst testing. Thermal cycling was then programmed into the SATS rig and consisted of first heating from room temperature (RT) to 300°C, then rapidly heating to 650°C, at which point IR furnace was turned off and cladding assembly allowed to air cool to 300°C, signaling one completed cycle and immediately ramping back up to 650°C. Once the thermal cycling regime was programmed, claddings were positioned within a quartz reaction tube environment and subjected to the simulated cyclic dryout. During testing, temperature was monitored at four different locations across the cladding, with a central thermocouple controlling the preprogrammed temperature regimes. 300°-650°C lifetimes for Zircaloy-2 varied from 1-4 cycles depending on internal pressure. During identical testing, C26M cladding did not fail. To generate failure criteria for C26M, it was necessary to increase cycling from 300°-650°C to 300°-700°/750°C. Actual cladding temperatures, for both Zircaloy-2 and C26M, were slightly higher than programmed temperatures due to thermal gradients within the furnace. Cyclic dryout maps illustrating the superiority of C26M over Zircaloy-2 regarding cyclic dryout lifetimes were drawn.

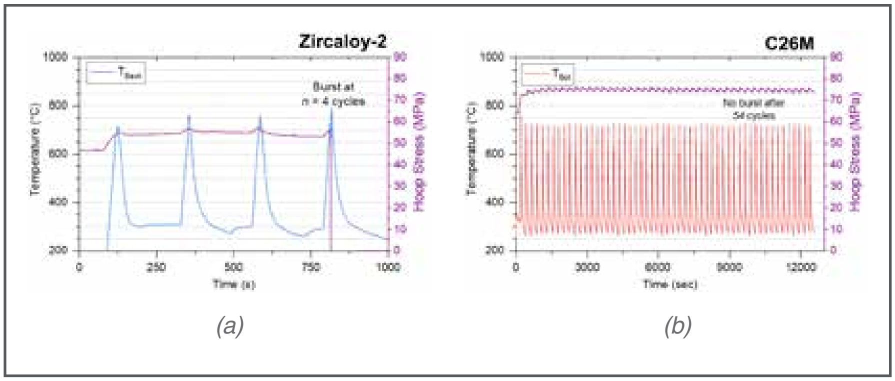

Figure 1. Boiling Transition (Dryout) Testing of FeCrAl Cladding (a) Temperature and hoop stress profile of Zircaloy-2 cladding during 300°-650°C temperature cycling at 55 MPa bursting at 4 cycles. (b) Temperature and hoop stress profile of C26M cladding during identical thermal cycling at a higher hoop stress. C26M did not burst after 54 cycles, at which point cycling was concluded.

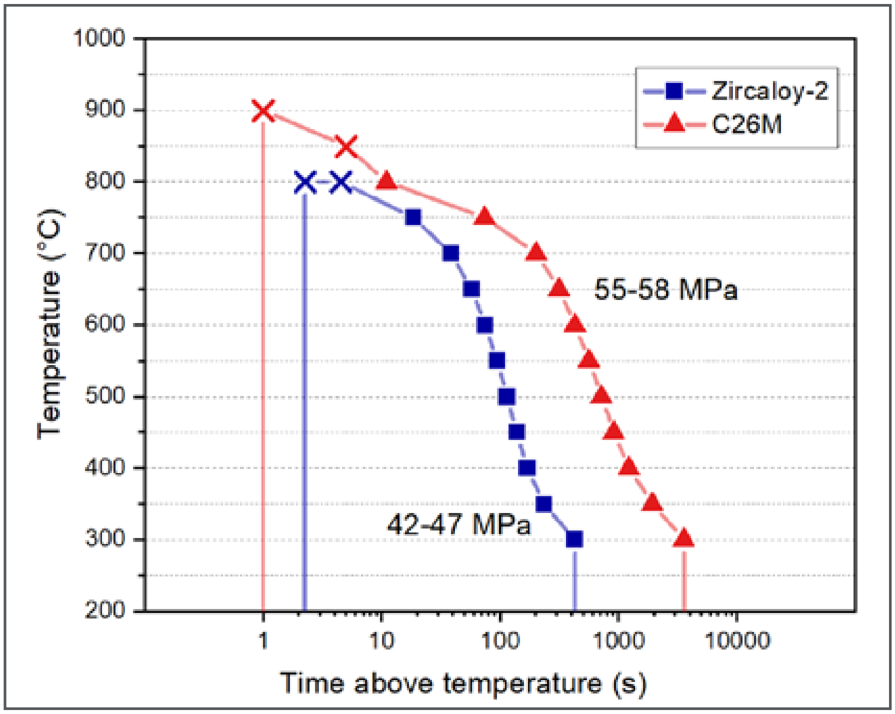

Figure 2. Boiling Transition (Dryout) Testing of FeCrAl. Dry out hoop stress map obtained by combining failure criteria from C26M burst at 55 and 58 MPa hoop stress cycled at 300°-700°C and 300°C-800°C, respectively, and Zircaloy-2 burst at 42 and 47 MPa hoop stress, cycled at 300°-650°C. The "X"s indicate when cladding failure occurred; the left most 'X' indicate single cycle cladding failure.

First Experiments and Evaluation of uranium silicide fuel and SiC Composite during an In-Pile Overpower Transient

Principal Investigator: David Kamerman
Team Members/ Collaborators: Nicolas Woolstenhulme, Devin Imhote, Austin Flemming, Colby Jensen, Charles Folsom, Connor Woolum, Korbin Tritthart, Jason Schulthess and Daniel Wachs

Integral testing of ATF concepts in RIA conditions is used to develop the regulatory safety criteria which govern their operation in commercial power plants.

Accident Tolerant Fuel (ATF) concepts include a wide variety of fuel and cladding materials, both as variants of the current Zircaloy-UO_2 system and also as novel fuel and cladding concepts such as Uranium Silicide (U_3Si_2) fuel and Silicon Carbide composite (SiC-SiC) cladding [1] [2]. These new materials will likely behave very differently in accident conditions requiring the development of new fuel safety criteria to regulate their performance. One of the most challenging design basis accidents (DBAs) for LWRs are rapid high-power excursions that are known as Reactivity Initiated Accidents (RIA). The first ATF concepts to be integrally testing in RIA conditions are U_3Si_2 in Zircaloy cladding and U_3Si_2 in SiC-SiC cladding. These tests were performed in the TREAT reactor. (50–150 words).

Project Description:

These first transient tests with ATF occurred in July of 2019 and involved the two rodlets with U_3Si_2 fuel and Zircaloy-4 cladding (SETH-F and SETH-G). The first rodlet (SETH-F) underwent two transients, the first of which was to confirm the expected core-to-specimen energy coupling factor (ECF) of 0.44 J/g-MJ (J/g of specimen energy per MJ of TREAT reactor energy). The purpose of the second test was to explore the behavior of U_3Si_2 fuel at an energy deposition slightly above the melting enthalpies. The SETH-G transient was tested at an even more aggressive transient to determine if changes in coolable geometry could be induced when pellet enthalpies significantly above the melting point occurred. The transients with the SiC-SiC clad rodlets (SETH-H and SETH-I) took place later in the fall of 2019 in October and November respectively. The purpose of SETH-H was to explore interactions of molten U_3Si_2 with SiC, and the purpose of SETH-I was to explore pellet cladding interactions between a still solid U_3Si_2 fuel pellet and SiC cladding. Cladding temperature during the transient was recorded using 4 thermocouples and two infrared pyrometers. It was also calculated using the BISON fuel performance code. Temperatures for the SETH-G transient are shown below in Measured and Calculated Temperatures During the SETH-G Transient. The SETH capsules

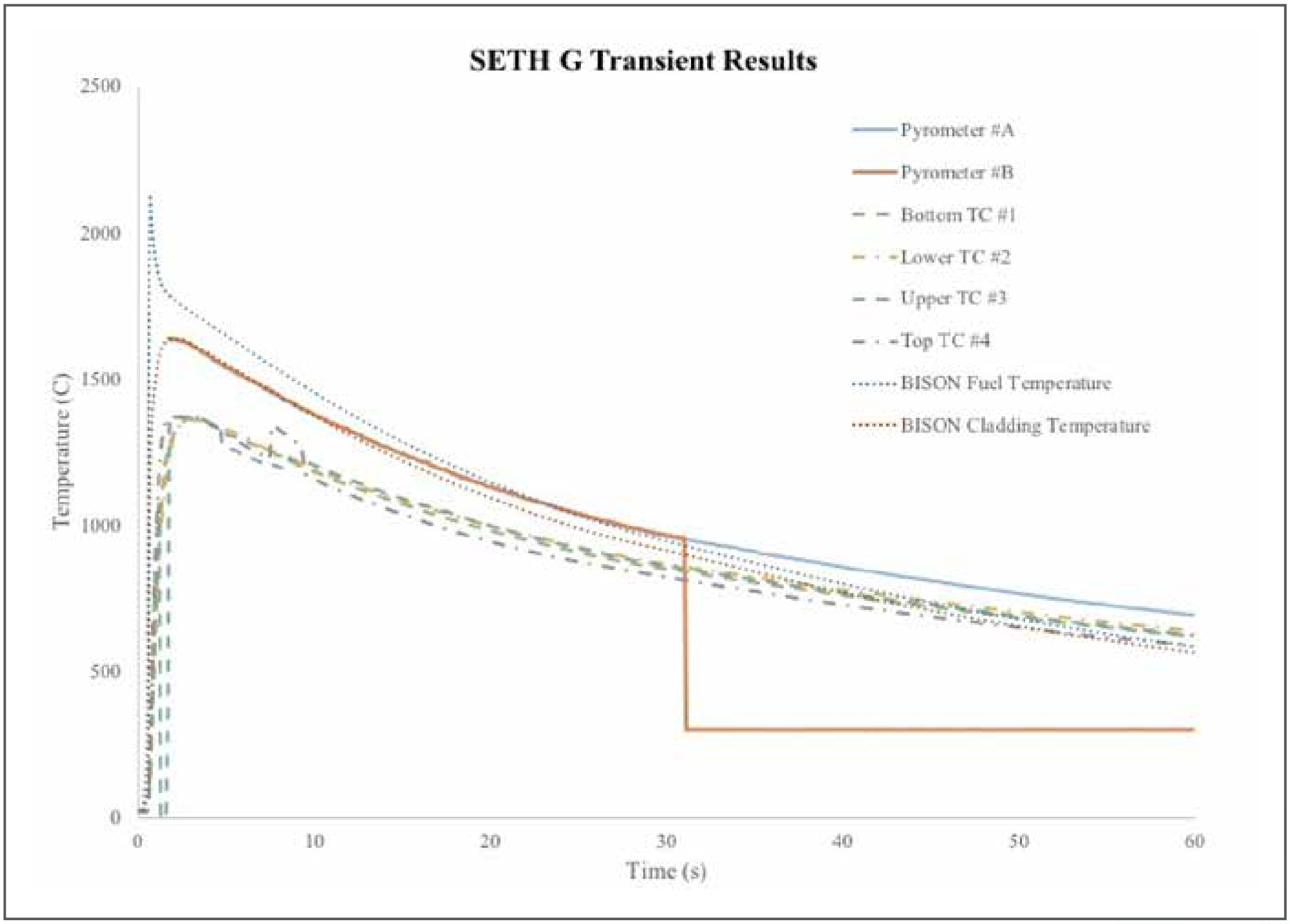

Figure 1. Measured and Calculated Temperatures During the SETH-G Transient.

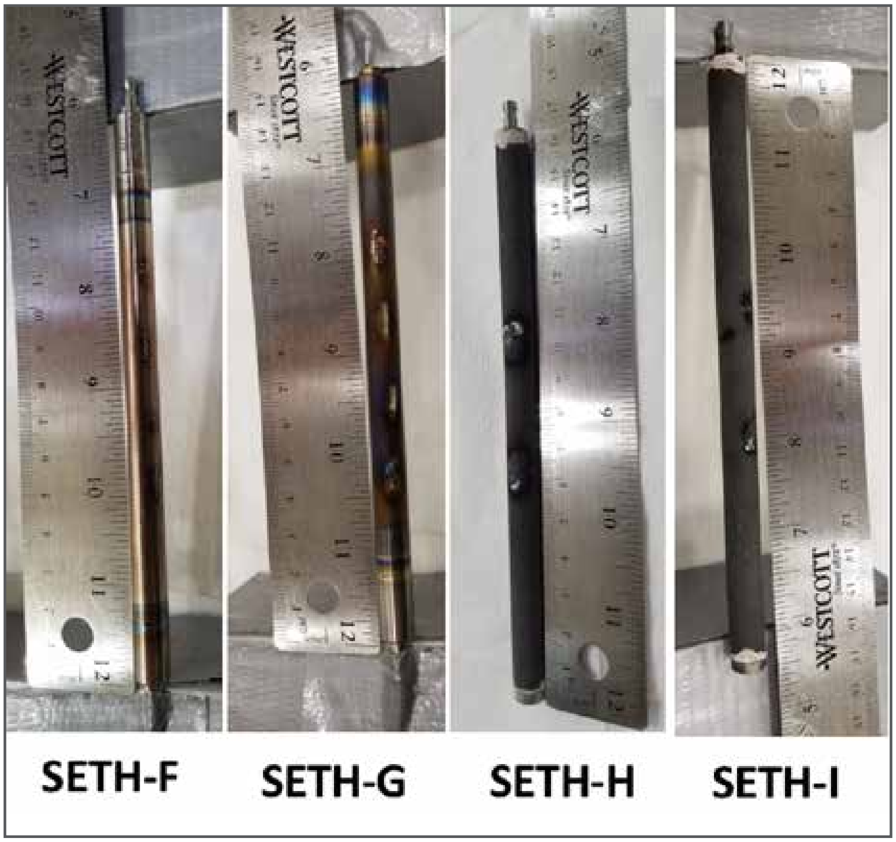

Figure 2. Seth F-I Post Transient Images.

containing the fuel test specimens were then transferred from TREAT to INL's Materials and Fuels Complex (MFC) for disassembly and post transient examination. Capsules were disassembled in a shielded nitrogen gas atmosphere glovebox to minimize oxidation and further degradation of the materials. Visual examination was performed through the glovebox wall/window with an optical camera with no image magnification. No gross changes in geometry were detected, however a likely fuel failure was seen in the SETH-I rodlet that will be further investigated.

Accomplishments:

Nuclear fuel safety criteria for RIA events are generally reported in terms of a peak radial average enthalpy or a change in peak radial average enthalpy. For the current LWR fuel design, the initial enthalpy limit for fuel rods consisting of UO_2 pellets incased in a Zircaloy cladding were established in 1974 and published in the Atomic Energy Commission (AEC) Regulatory Guide 1.77 [3]. Initially, a limit was set to 1170 J/g UO_2 radial average fuel enthalpy for both fresh and irradiated fuel based on the maintenance of rod-like (coolable) geometry. This enthalpy limit was based on a review of Special Power Excursion Reactor Test (SPERT) and TREAT experimental data from the 1960's. Leading on from the PBF experimental program, testing of higher burnup fuels in the 1980's and 1990's performed in Russia, Japan, and France began to uncover PCMI as a new fuel failure mechanism for fuels with extended burnups. The current RIA fuel failure limits are inclusive of low temperature failures (PCMI driven) and high temperature failures (swelling and rupture driven). The high temperature failure limit starts at 711 J/g and decreases to 628 J/g with increasing fuel rod internal pressure. The low temperature (PCMI) limit is based on the state of environmental degradation in the cladding and starts at 628 J/g and decrease to 209 J/g [4]. Based on these initial experiments, ATF rods that consist of U_3Si_2 fuel in Zircaloy cladding or SiC-SiC cladding, fuel safety limits could start as high as 528 J/g for fresh fuel based on the results of the experiments described above (SETH-G). While this is lower than the current limits for fresh fuel (628 J/g or 711 J/g), staying under this value should not pose an excessive burden on LWR reactivity control systems. These limits will likely need to be reexamined for the fuel as its burnup increases. The SETH-I test provides some insight into how this limit may change. The SETH-I test simulates pellet swelling during burnup and closure of the pellet cladding gap. The SETH-I test showed that for fuel rods with U_3Si_2 fuel and SiC-SiC cladding the fuel rod is likely able to retain its geometry at enthalpies up to 330 J/g even when PCI occurs.

References:

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[3.] U.S. Atomic Energy Commission, "Regulatory Guide 1.77," no. May, pp. 1–8, (1974)

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First Integral RIA Experiments in the MARCH-SERTTA Static Water Capsule

Principal Investigator: Charles Folsom
Team Members/ Collaborators: Nicolas Woolstenhulme, Colby Jensen, David Kamerman, Daniel Wachs, Devin Imholte, Leigh Ann Emerson, Andrew Chipman, Robert Armstrong and Spencer Snow (INL)

Figure 1. Disassembly of ATF-RIA-1-Gamma capsule in TREAT glovebox to send rodlet for gamma spectroscopy

The restart of the Transient Reactor Test (TREAT) facility has renewed the capability for transient testing of nuclear fuels in the United States. This capability is critical to support the mission of the Accident Tolerant Fuels (ATF) campaign to develop the next generation of fuels with enhanced tolerance to design basis accidents (DBAs) such as a reactivity-initiated accident (RIA). The design and development of the Minimal Activation Retrievable Capsule Holder–Static Environment Rodlet Transient Test Apparatus (MARCH-SERTTA) provides the capability to test Light Water Reactor (LWR) fuel rods in a static water environment under RIA-like conditions. The ATF-RIA-1 test campaign includes a series of six tests to commission the MARCH-SERTTA capsule for RIA testing in TREAT.

Project Description:

The Minimal Activation Retrieval Capsule Holder-Static Environment Rodlet Transient Test Appartatus (MARCH-SERTTA) capsule provides a static water environment surrounding

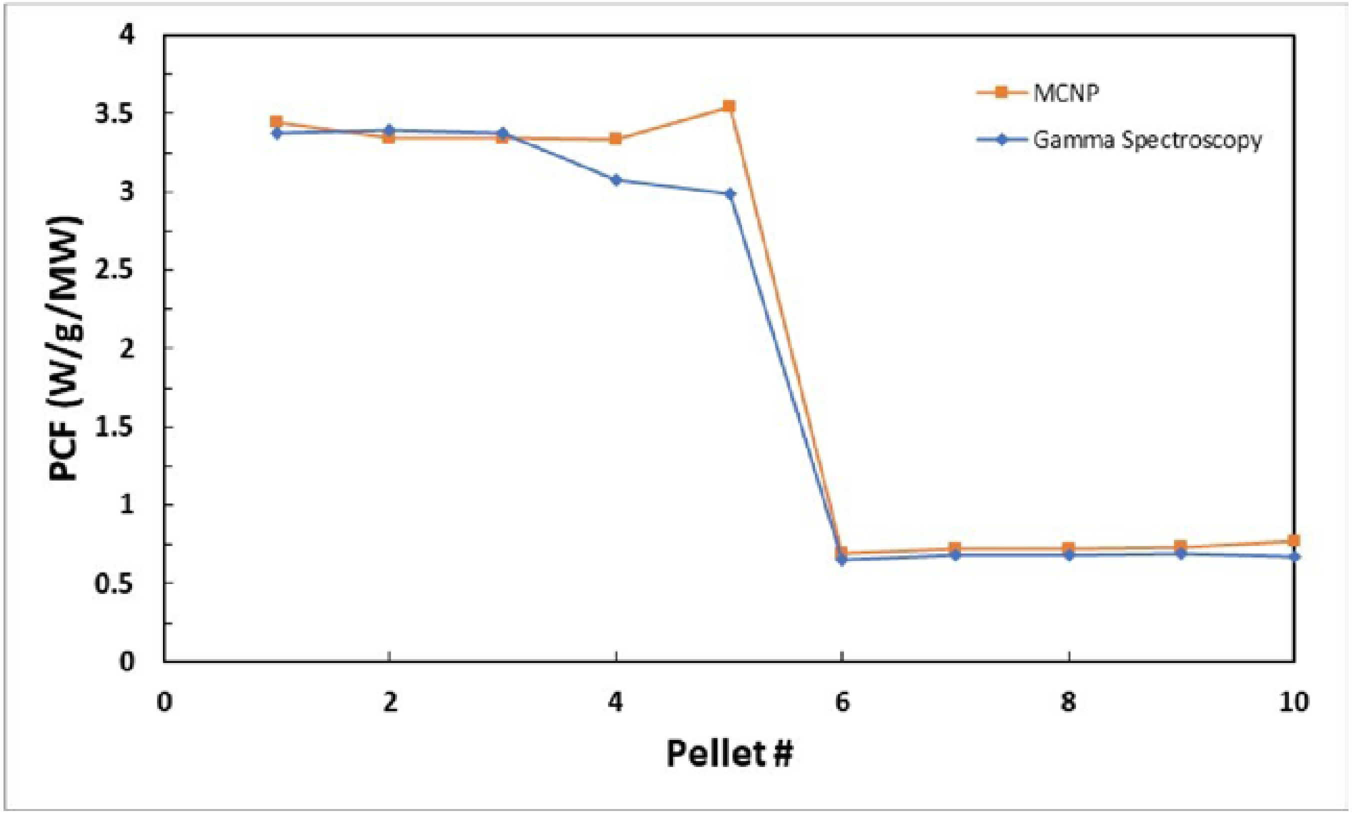


Figure 2. MCNP predictions and gamma spectroscopy measurements of the core-to-specimen power coupling factor of ATF-RIA-1-Gamma rodlet with five 4.95wt% and five 0.74wt% U^{235} fuel pellets. Average measured to predicted deviation was less than 6%.

the fuel rod with a suite of state of the art and first of a kind instruments to collect relevant data for post-test analysis of the experiments. The instruments include thermocouples and optical-fiber-coupled pyrometers to measure the temperature of cladding and water, a pressure transducer to monitor capsule pressure, an electro-impedance detector to monitor boiling around the rod, and linear variable differential transformer (LVDT) to monitor axial elongation of the fuel rod and pressure in the rodlet plenum. The purpose of the ATF-RIA-1 tests is to commission the MARCH-SERTTA capsule and instrumentation for RIA testing in a static water environment in TREAT. Demonstrating and proving the capabilities of the MARCH-SERTTA capsule is crucial for planned follow-on testing of ATF as well as very high burnup LWR fuels. These tests will play a crucial role in meeting industries needs to support licensing these fuels.

Figure 3. Assembly of ATF-RIA-1-A capsule.

The ATF-RIA-1 commissioning series includes six tests. The first test (ATF-RIA-1-Gamma) was a calibration test to determine the relationship between the energy generated in TREAT versus the energy generated within the rodlet fuel. The rodlet was constructed with fuel pellets of both 4.95wt% and 0.74wt% U^{235} so that an upper and lower bound could be determined for the core-to-specimen energy coupling. Gamma spectroscopy was performed on this rodlet to provide a measured quantity for the core-to-specimen energy coupling. The remaining ATF-RIA-1-A through E tests are planned with varying specimen energy deposition targets from ~500-1100 J/gUO_2 and capsule initial temperatures from 20°C up to 200°C. These conditions will provide baseline assessment of fresh UO_2-Zry fuel rods covering regulatory limits for energy deposition during RIA, with varying anticipated failure mechanisms from no failure expected; to possible high temperature embrittlement failure of the cladding or ballooning and burst of the cladding.

Accomplishments:

During fiscal year 2020, the MARCH-SERTTA experiment campaign completed the ATF-RIA-1-Gamma test in January of 2020. The capsule was disassembled in a glovebox at TREAT and the rodlet was sent for gamma spectroscopy. The gamma spectroscopy provided a measured result of the number of fissions that occurred during the test to calculate the TREAT core-to-specimen energy coupling. The measured and predicted results from the neutronic calculations using MCNP show very good agree-

Assessment of the mechanical properties of high burnup cladding is essential to establish a licensing criteria for modern and advanced nuclear fuel designs in light-water reactors.

ment with an average deviation less than 6%. This provides a high level of confidence that the planned transients will achieve the targeted energy depositions for the remaining tests in the ATF-RIA-1 series.

In July 2020 the MARCH-SERTTA team successfully completed the first high energy RIA test in a water environment to support LWR fuel safety testing in TREAT since the 1960's and the first integral test in the U.S. since the 1970's. The ATF-RIA-1-A capsule was irradiated with a 4.2% Δk/k TREAT transient clipped to 1260 MJ that deposited 860 J/gUO_2 into the fuel specimen in a 90.5 millisecond full-width-half-max pulse. The results showed cladding temperatures measured in excess of 1200°C. Disassembly and inspection of the capsule and rodlet is still pending. The ATF-RIA-1-B and C capsules have been assembled and are currently awaiting irradiation at TREAT and capsuled D and E are undergoing assembly and are scheduled to be irradiated in TREAT during September 2020. These experiments will complete the commissioning the MARCH-SERTTA capsule for integral RIA testing in TREAT paving the way to supporting ATF and high burnup testing needs. These experiments provide the foundation to supporting a wide range of already planned experimental programs in coming years such as the Nuclear Energy Agency (NEA) FIDES project HERA, Westinghouse ATF, Framatome ATF, General Electric ATF, and Nuclear Science User Facilities/Nuclear Energy University Project (NSUF/NEUP) project for coated claddings.

Figure 4. Rodlet holder and instrumentation around rodlet of a partially assembled ATF-RIA-1 capsule shown next to an enhanced x-ray radiograph of ATF-RIA-1-A capsule prior to irradiation (image courtesy of David Chichester).

Experiment Design for Prototypic In-Pile LOCA Evaluations

Principal Investigator: Nicolas Woolstenhulme
Team Members/ Collaborators: Colby Jensen, Charlie Folsom, Robert Armstrong, Dan Wachs and Guillaume Mignot (OSU)

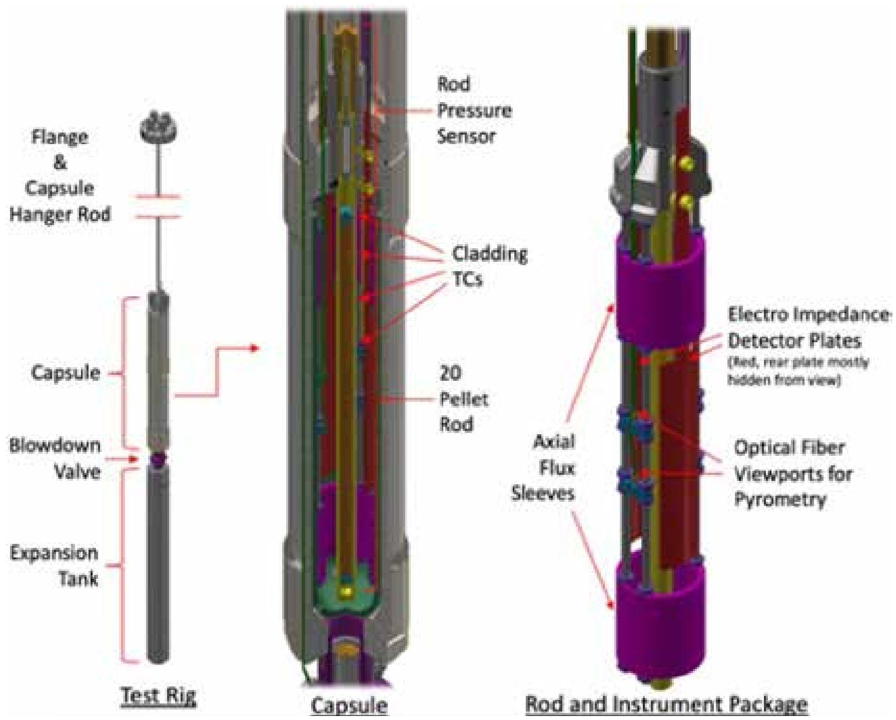


Figure 1. LOCA-SERTTA Design Rendering.

In 2018 the Halden Boiling Water Reactor (HBWR) unexpectedly announced termination of reactor operations and its long-standing irradiation programs. Among other things, this event represented the loss of the western world's only remaining nuclear-heated capability for testing postulated Loss of Coolant Accident (LOCA) conditions in Light Water Reactors (LWR). At the same time there was a general resurgence of interest in extending the licensed burnup limits for LWR fuel with specific budget scope starting in 2020 to address relevant data gaps. One major research area, referred to as Fuel Fragmentation Relocation and Dispersal (FFRD), had been identified but not fully resolved in past HBWR LOCA tests on high burnup LWR fuel. Efforts were undertaken in fiscal year 2020 to evaluate candidate methods for performing LOCA tests

The capsule-based approach to LOCA testing at TREAT will help close near-term data gaps for in-pile fuel fragmentation phenomena to support licensed burnup increases for LWR fuel technologies.

in the recently restarted Transient Reactor Test facility (TREAT). Thermal-hydraulic modeling was performed on three candidate test vehicle designs to determine which represented the best approach for resolving FFRD-related data gaps in support of near-term project goals. This summary gives a brief overview of these efforts and conclusions.

Project Description:

Three irradiation vehicle approaches were conceptualized for TREAT-based LOCA testing. These devices were compared to each other with thermal hydraulic modeling in the RELAP code. The first was a small capsule in the Static Environment Rodlet Transient Test Apparatus (SERTTA) family of water capsules entitled LOCA-SERTTA. The second was an enlarged capsule version referred to as Super-SERTTA. The third was an even larger pump-driven system termed the TREAT Water Environment Recirculating Loop (TWERL). All three concepts had a blowdown tank connected via remote-operable valve so that TREAT's reactor power history could be synchronized with capsule depressurization to faithfully simulate the postulated transition from steady state operation to LOCA conditions. Unlike furnace-based LOCA testing where external heating is used (which is a valid approach for LOCA testing centered around cladding performance) TREAT's ability to drive internal nuclear heating offered the unique ability to simulate representative radial power profiles and resulting thermomechanical stresses in the test fuel. Hence, design tuning and modeling emphasis was placed on simulating the full evolution of both cladding and fuel centerline temperatures throughout the LOCA evolution.

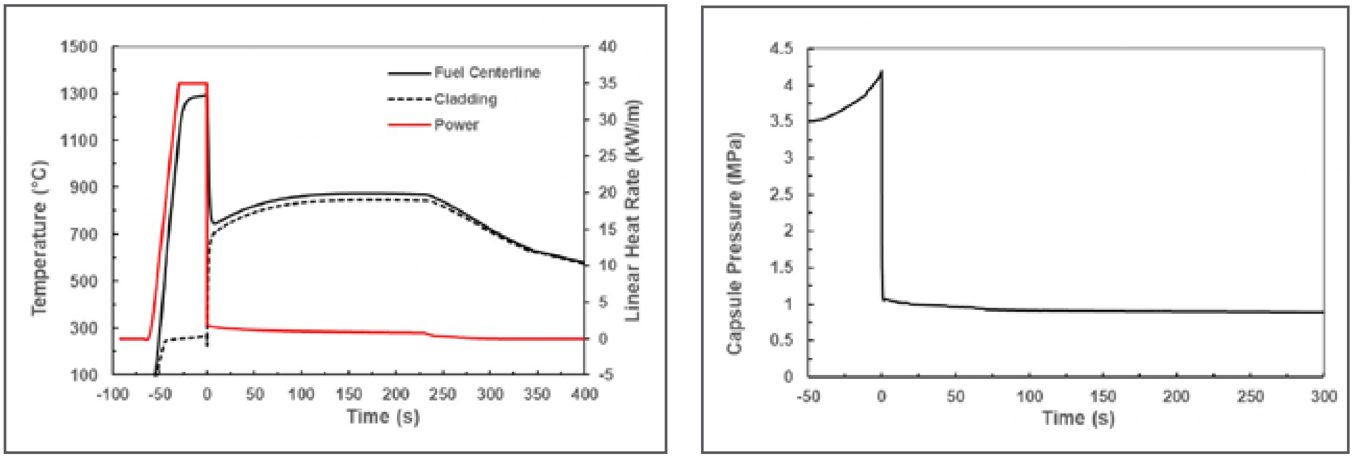


Figure 2. RELAP Predictions of LOCA-SERTTA Testing Method.

These predictions were compared to the same metrics obtained from a RELAP model of a typical pressurized water reactor plant.

Accomplishments:

The design and analysis team iterated on several engineering and operating parameters in comparing these three candidate irradiation devices. LOCA-SERTTA was found to be remarkably effective in simulating the pre-blowdown conditions by using a brief period of nucleate boiling from the rod to enhance heat transfer. Super-SERTTA was found to be effective in this regard by using a tall column of water aided by natural circulation. TWERL was naturally found to produce the most representative pre-blowdown heat transfer conditions due to pump-driven forced circulation. All three devices were found to offer adequate post blowdown pressure to permit cladding differential pressure to drive ballooning phenomena.

LOCA-SERTTA's use of a 20 pellet rodlet and small steam suppression pool at the bottom its expansion tank enabled adequate performance in a compact geometric envelope able to be fit entirely within existing safety containment structures at TREAT; a major boon for the prospect of near-term cost-effective testing. Super-SERTTA's tall water column was shown to significantly quench the fuel rod as water rushed out during blowdown and created challenges with representing the desired temperature response.

Super-SERTTA, however, did offer the ability to test longer rods in a capsule device. TWERL was found to represent the desired conditions very well and offered the notable ability to house longer rods in multi-rod bundles to help simulate refined post blowdown boundary conditions affected by neighboring rods such as azimuthal temperature variations and balloon-to-balloon interactions. The ultimate recommendation reached by these studies was to focus on detailed design and deployment of LOCA-SERTTA to address near term FFRD data needs for baseline UO2 fuel in zirconium alloy cladding. The TWERL concept was also recommended for continued pursuit for its ability to address all relevant phenomena for advanced accident tolerant fuel (ATF) designs so that it is ready when high burnup ATF specimens become available.

Having reached this conclusion, the team proceeded to perform detailed design work and started mockup testing of LOCA-SERTTA features. Experiment safety analyses were largely performed to permit its irradiation at TREAT. Finally, collaborators at Oregon State University (OSU) created mock-ups of the LOCA-SERTTA capacitive discharge burst disc rupturing system and demonstrated it viability as a compact remote operable valve to induce the blowdown. The team held both conceptual and preliminary design reviews in accordance with INL engineering procedures. The device was designed with an instrumentation package consistent with, and in some regards advanced beyond, the approach used at the HBWR. The capability to represent fully prototypic thermal evolution in the fuel will be unique in the world and beyond the in-pile capability once available at HBWR, as a crucial capability to address FFRD related questions for high burnup fuels in the near term and to support detailed characterization of ATF fuels as well. Future work in 2021 will complete the final design phase and construct a full assembly prototype for lab testing in preparation for commissioning irradiations with fresh fuel later in the year.

Non-Destructive Post irradiation Examination Results of the First Modern Fueled Experiment in TREAT

Principal Investigator: Jason Schulthess
Team Members/ Collaborators: Nicolas Woolstenhulme, Aaron Craft, Joshua Kane, Nicholas Boulton, William Chuirazzi, Alexander Winston, Andrew Smolinski, Colby Jensen, David Kamerman and Daniel Wachs

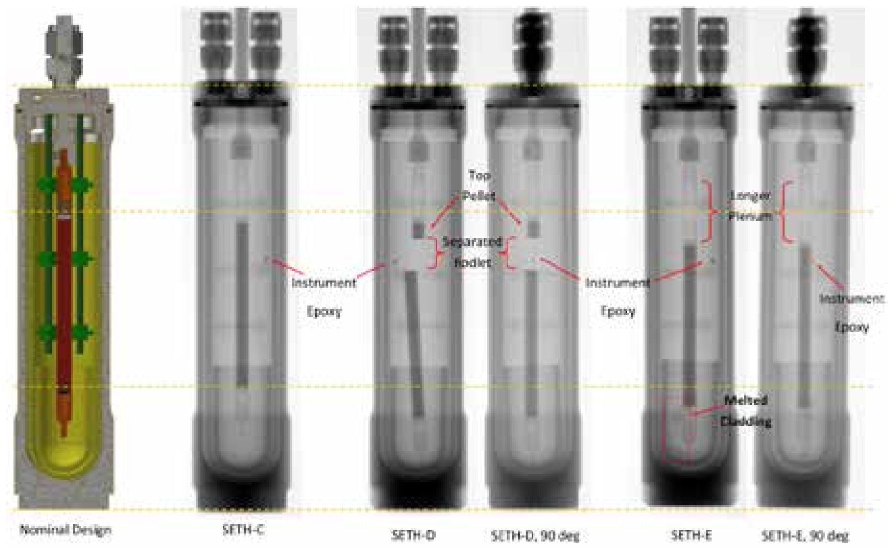


Figure 1. Post Transient Neutron Radiographs of SETH-C, D, and E obtained at the NR station at TREAT.

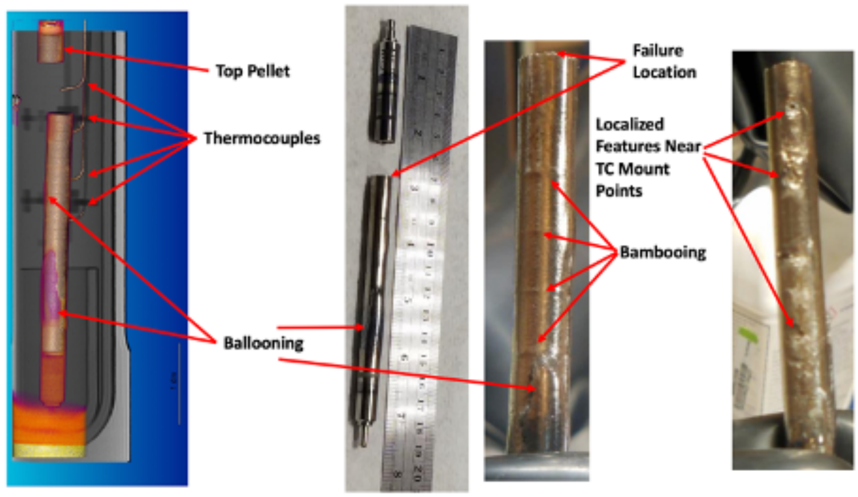
The restart of Transient Reactor Test (TREAT) has enabled a modern era of fuel-safety research testing to begin. The unique configuration of TREAT enables test designs to perform science-based phenomenological separate-effects investigations, integral-scale fuel testing for water-cooled reactors, or loop testing capabilities for advanced reactors. In each case, the tests can be highly instrumented with state-of-the-art in situ instruments not previously available in the prior era of fuel-safety research testing.

As with advances implemented into the restarted TREAT reactor, meaningful advances in post irradiation examination (PIE) capability have also occurred in the time since fuel-safety research tests were last conducted. This paper will focus on the advanced nondestructive post-irradiation examination capabilities that have been developed and their use to conduct the first fueled tests performed in TREAT in the modern era. These examinations will include gamma emission tomography, digital neutron computed tomography, and visual examinations.

Demonstration of irradiation and non-destructive post-irradiation examinations using modern non-destructive techniques provides a baseline set of data on known materials and opens the door to performing extreme nuclear heated tests on advanced materials including accident tolerant fuel materials.

Project Description:
A series of experiments was conducted in the TREAT Facility as part of commissioning for nuclear fuel safety research to be continued over the next several decades. TREAT was first constructed in the late 1950s to support research on nuclear fuel specimens under extreme nuclear-heated conditions. Following a decades-long hiatus, reactor operations were resumed at TREAT in 2017 to support the reemerging field of fuel-safety research. The first fueled tests were performed on fresh light-water reactor (LWR) type sub length specimens (4.9% enriched UO_2 in zirconium-alloy cladding) in inert gas capsules with the primary objective to demonstrate new irradiation capabilities, including calculation of energy-coupling factor (ECF), and to provide a baseline reference for interpretation of future experiments on accident tolerant fuel (ATF) designs and high burnup standard fuels. In parallel, several new nondestructive post irradiation experimental systems and processes have been developed and were successfully demonstrated using these samples. Five capsules were irradiated with energy injections and peak cladding temperatures of between ~212 and ~1312 J/gUO_2 and ~561–~2113°C, respectively. These irradiations resulted in various degrees of fuel damage that were characterized using advanced non-destructive examinations. The PIEs highlighted that, in the most-energetic transient, zirconium breakout, also referred to as 'candling,' was observed as gross cladding melting and relocation. PIEs also confirmed results inferred from in situ instrumentation during irradiation and are consistent with previous results from other studies.

Figure 2. 3D nCT renderings of SETH-D and initial visual inspections showing regions of ballooned cladding and localized features such as pitting near thermocouple mounting points. The false colors shown in the nCT rendering have no scale or specific quantitative value but are applied to highlight finer features in the tomography which otherwise are difficult to visualize.



Accomplishments:

TREAT's first year of experimental operations since 1994 was focused on demonstrating new LWR relevant transient shapes representing Reactivity-Initiated Accidents (RIAs) and Loss of Coolant Accidents (LOCAs). The successful outcome of these efforts paved the way for the first fueled experiments using a new irradiation vehicle system, associated instrumentation, including novel application for pyrometry and other support systems which were successfully commissioned. These efforts were paralleled by development of advanced non destructive post-irradiation examination technique development of Gamma Emission Tomography (GET) and digital neutron-computed tomography (nCT) that were specifically focused on assessment of complex and disrupted fuel samples. While GET shows promise, additional development is needed to increase spatial resolution while at the same time reducing counting time for low source-strength specimens. The commissioning demonstration of the first digital neutron computed tomography on irradiated fuel has proven successful and valuable for both its speed and resolution of data acquisition.

In addition to demonstrating and commissioning the post irradiation examination capability, the post irradiation exams confirmed the results of the predicted irradiation conditions and in situ instrumentation—namely, that no cladding melting would occur in Separate Effects Test Holder (SETH) C, that SETH D was very near the cladding melt temperature, and that

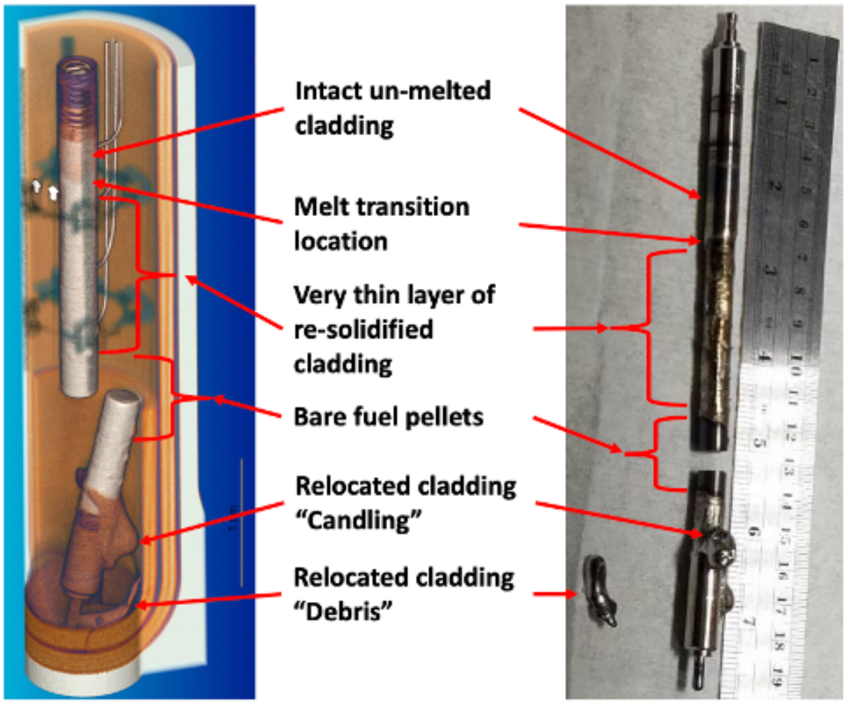


Figure 3. 3D nCT renderings and visual examination of SETH-E showing the disrupted fuel rodlet, relocation of melted cladding, and droplets of cladding material down the sides of the pellets (i.e. candling). Temperature transition regions are apparent, where cladding exceeded or stayed below the melting temperature. As previously explained, the rendering contains several periodic artifacts as a result of the rendering engine used. The false colors shown in the nCT rendering have no scale or specific quantitative value but are applied to highlight finer features in the tomography which otherwise are difficult to visualize.

SETH-E well exceeded the cladding melt temperature resulting in gross cladding relocation. The difference in gross cladding relocation was observed to be sensitive to temperature-driven viscosity and the time above the cladding melt temperature, both of which enhance viscous flow.

Finally, the use of separate effects tests using the SETH capsule-irradiation device was shown to be viable and the correct qualitative failure mechanisms were predicted, demonstrated, and observed. The combination of TREAT and the Minimal Activation Retrievable Capsule Holder (MARCH) irradiation device are considered ready for use in future fuel safety research applications.

Future work on the material from this test series includes performing destructive characterization such as optical microscopy with the intent to identify fuel cladding interaction, cladding phase changes, melt/re-freeze fonts, voiding, cladding cracks, etc. It is intended to make this additional information available once this work has been completed.

2.6 LWR COMPUTATIONAL ANALYSIS

Reactor Performance and Safety Impacts of Increased Enrichment

Principal Investigator: Nicholas R. Brown
Team Members/ Collaborators: Joseph R. Burns, Richard Hernandez, Kurt A. Terrani and Andrew T. Nelson

The research analysis indicated that increasing fuel enrichment had no detrimental impacts on reactivity coefficients, and the rim effect impacts on the distributions of Pu content, EOL burnup and HBS, as well as fission gas are enhanced with increasing enrichment.

Advanced Fuels Campaign (AFC) is developing Accident Tolerant Fuel (ATF) materials capable of increasing the safety margins of light water reactors (LWRs) under design basis and beyond basis accident conditions [1]. These materials offer enhanced reactor performance under nominal conditions due to reduced hydrogen pickup in the advanced cladding, which leads to the potential for higher achievable burnup (with concomitant enrichment increase), longer fuel cycle length, and better fuel utilization. We aimed to quantify the reactor performance, safety impacts, and fuel cycle implications of increasing pressurized water reactor (PWR) 235U fuel enrichment beyond 5%.

Project Description:

The technical objective of this research was to quantify high-level observations on the reactor performance and safety impacts, as well as several fuel cycle metrics associated with implementing PWR fuel enrichments between 5-7% 235U. This research is helping the nuclear industry reduce costs for the nation's current and next generation reactors by leveraging the economic advantages of High Assay Low-Enriched Uranium (HALEU) fueled LWR systems, mainly higher fuel utilization, while ensuring that HALEU fuel in LWRs does not introduce added safety and reliability concerns throughout their operational lifetime. To achieve this, several high-level performance indicators used in the research and applications of pin-level lattice evaluations assessed several performance metrics. Adding to the AFC knowledge base, discharge burnup was the main reactor performance metric used to assess the economic advantages of HALEU fuel, coupled with involving several fuel cycle evaluation and screening (E&S) evaluation criteria (natural resource utilization, spent fuel metrics, environmental impacts) to characterized the economic impacts of increasing PWR fuel enrichment [3]. To evaluate the safety performance of HALEU fueled LWRs, reactivity coefficients (fuel and moderator temperature coefficients, soluble boron coefficient) were evaluated as a function of fuel enrichment. The benefits of the research can also add to the state-of-knowledge by addressing issues associated with boron control mechanisms, the effects on formation of high burnup structures (HBS), as well as radial fission gas and plutonium isotopes contents as a result of incorporating HALEU fuel cycles in PWRs.

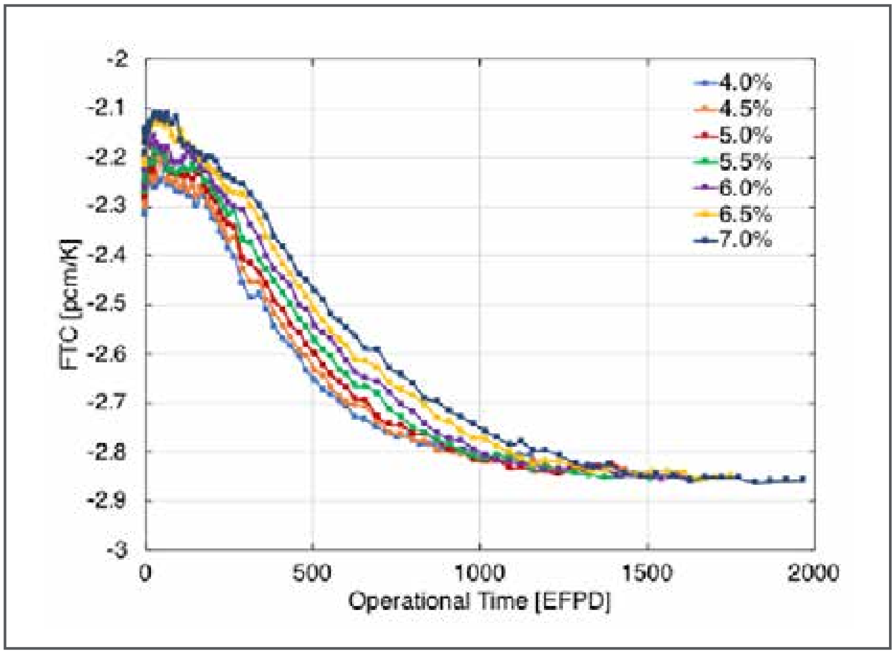


Figure 1. FTC variation versus cycle length for all enrichment cases.

Accomplishments:
Significant accomplishments were made towards several technical goals as part of this research involved with increased enrichment of PWR fuel above 5%. We performed pin-cell level lattice calculations using the SCALE code to accomplish an investigation of the fuel cycle length, beginning-of-life (BOL) and end-of-life (EOL) neutron spectra, reactivity coefficients, as well as the fuel burnup, the fission gas (krypton and xenon) and Pu content EOL distributions for seven different PWR fuel enrichment cases between 4-7% 235U. The fuel cycle discharge burnup was modeled as calculated from a 3-batch linear reactivity model assumption, with 3% leakage, and results showed a 10.9 MWd/kgU/% slope with enrichment. Further, increasing enrichment resulted in a hardening BOL and EOL neutron spectrum and the reactivity coefficients were shown to remain comfortably negative (Figure 1) for all the enrichment cases considered. Additional contributions included a comparison

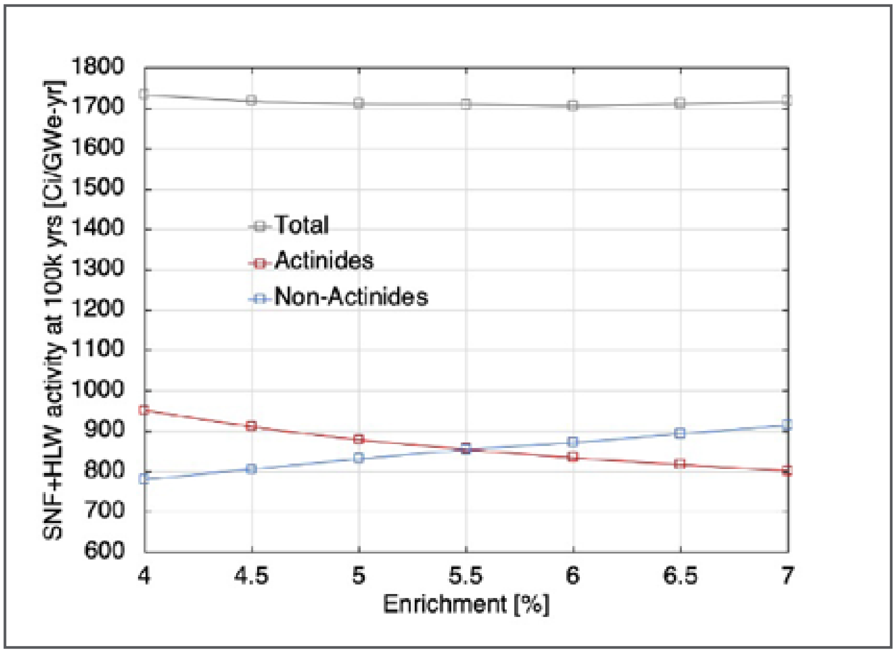


Figure 2. SNF+HLW activity contributions at 100,000 years after fuel discharge versus enrichment..

of these pin-cell level reactivity coefficients with those from the AP-1000 design control document (DCD) full core data, both showing comparable similarities. We also performed a fuel cycle performance analysis of the considered enrichment cases. This fuel cycle evaluation included evaluating the natural resource utilization, nuclear waste management, and environmental impacts associated with operating these HALEU fuels in PWRs. The general takeaway from this fuel cycle evaluation was that resource utilization and environmental impacts were similar for all the enrichment cases when normalized per GWe-year of energy. Further, nuclear waste metrics (Figure 2) showed that HALEU had similar spent fuel activity at 100 and 100,000 years after fuel discharge as that of fuel enrichments below 5% 235U. Richard's contributions also included a discussion of burnable absorber incorporation to control the expected BOL excess reactivity of HALEU to the state-of-knowledge of the AFC. We used a simple approach to quantify the susceptibility to pulverization (associated with HBS formation at burnup>71 MWd/KgU) [4] under accident scenarios with increasing enrichment for these PWR HALEU fuel cycles. These contributions (Figure 3) analyzes the fraction of pellet volume susceptible for pulverization as a function of enrichment, and conservative results showed that this becomes an increasing issue with increasing enrichment.

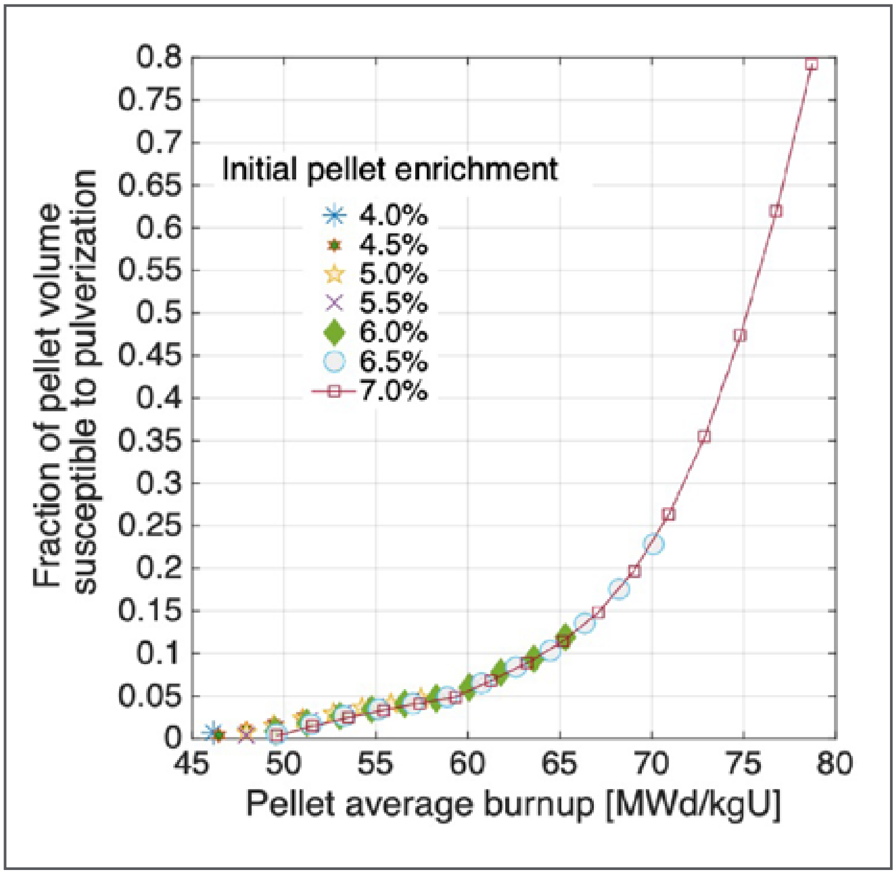


Figure 3. Volume fraction of the fuel restructured and susceptible to pulverization as a function of pellet average burnup.

References:

[1.] Terrani, K. A. "Accident Tolerant Fuel Cladding Development: Promise, Status, and Challenges." Journal of Nuclear Materials 501 (2018) 13–30.

[2.] Hernandez, R.; Todosow, M; Brown, N. R. "Micro heat pipe nuclear reactor concepts: Analysis of fuel cycle performance and environmental impacts." Annals of Nuclear Energy 126 (2019) 419-426.

[3.] Wigeland, R., et al., 2014. "Nuclear fuel cycle evaluation and screening—final report," INL/EXT-14-31465 FCRD- FCO-2014-000106 (Idaho Falls, ID, Idaho National Laboratory, October 8, 2014).

[4.] Turnbull, J.A.; Yagnik, S.K.; Hirai, M.; Staicu, D.M.; Walker, C.T. "An assessment of the fuel pulverization threshold during LOCA-type temperature transients," Nucl. Sci. Eng. 179 (2015) 477–485.

Impacts on LWR Reactor Performance and Safety Characteristics of Coated UN Fuel

Principal Investigator: Michael Todosow
Team Members/ Collaborators: Arantzazu Cuadra

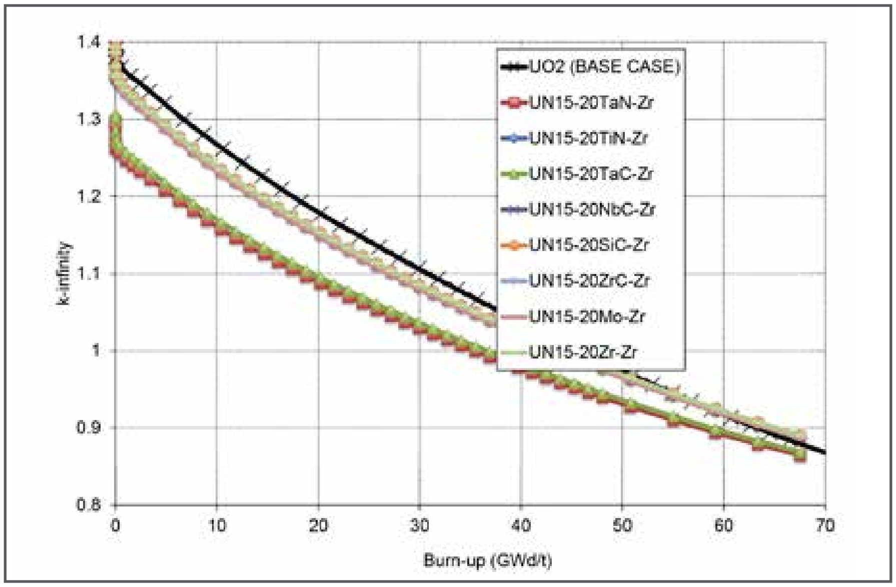


Figure 1. k-infinity vs. burnup.

In order to utilize UN fuel in commercial LWRs they must be protected to mitigate potential fuel/ coolant interactions. This study performed an initial assessment of the impacts on reactor performance and safety characteristics of 20 µm of several potential coatings on the UN fuel pellets. Knowledge of these impacts helps to inform the selection of potential coatings.

Following the catastrophic accident at the Fukushima Daiichi reactors in 2011, major efforts have been undertaken to develop fuels and claddings with enhanced accident tolerance (aka, Accident Tolerant Fuels – ATF). An ATF that is being pursued by Westinghouse is uranium-nitride (UN) with fully enriched N-15 to mitigate the absorption penalty of N-14. However, UN adversely interacts with the coolant in LWRs, and this study examined the impact on reactor performance and safety characteristics of coating UN fuel pellets to mitigate the fuel/coolant interactions in LWRs. Several candidates proposed by LANL for the coatings were analyzed. Initial studies assumed 20 µm thickness for the coatings applied to the surface of the pellet.

Project Description:

Scoping calculations were performed to estimate the impact on reactor performance and safety characteristics of coating UN fuel pellets to mitigate the fuel/coolant interactions in LWRs. The analyses were performed with the TRITON lattice code for a detailed model of a reference 17x17 Westinghouse fuel assembly and the fuel was assumed to be fully enriched N-15. Initial studies assumed 20 µm thickness for candidate coatings proposed by LANL. The coatings were applied to the surface of the pellet and the fuel pellet outer radius and cladding radii were fixed at the reference values to facilitate insertion in existing PWRs. Cycle length, discharge burnup and

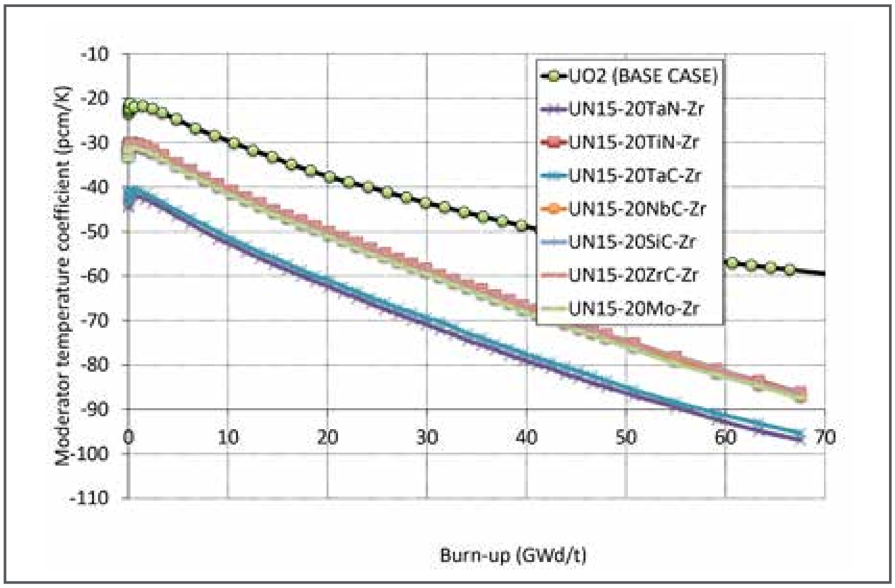


Figure 2. Moderator Temperature Reactivity Coefficient vs. burnup.

fuel and moderator temperature reactivity coefficients, and soluble boron and control worth were determined as functions of burnup.

Accomplishments:

Uranium-nitride (UN) has several characteristics the make it an attractive fuel for utilization in existing commercial LWRs; these include higher density and thermal conductivity than traditional UO_2 fuel. These characteristics can reduce the enrichment required to achieve desired cycle length and discharge burnup, and improved reactor performance in normal and accident conditions. However as noted above, UN reacts adversely with light-water and therefore must be protected in the event of a cladding breach. Coatings of several candidate materials proposed by LANL were evaluated to provide and initial estimate of their impacts on reactor performance and safety characteristics: TaN, TaNi, TaC, NbC, SiC, ZrC, Mo, Zr .

The major observations can be summarized as:

- There is no difference between these coatings for the fuel temperature and soluble boron reactivity coefficients vs. burnup, but both are less negative than for UO_2 reflecting the harder spectrum from UN
- For K-infinity and MTC, most coatings exhibit a similar and stronger "negative" than for UO_2; this is especially pronounced for TaN and TaC. Results for both k-infinity and MTC are shown in Figure 1 and 2 respectively.

The cycle length and discharge burnup based on the Linear-Reactivity-Model (LRM), and 3-batch fuel management for the non-Ta coatings were approximately 700 days/60 GWd/t, while for the Ta-based coatings they were approximately 550 days/46 GWd/t. The corresponding values for UO_2 are ~530 days/61 GWd/t.

Transient/accident analyses will need to be performed to quantify the impacts of the observed differences in reactivity and control coefficients.

Neutronic Analyses to Support Accelerated Testing of LWR Fuels

Principal Investigator: Michael Todosow
Team Members/ Collaborators: Arantzazu Cuadra

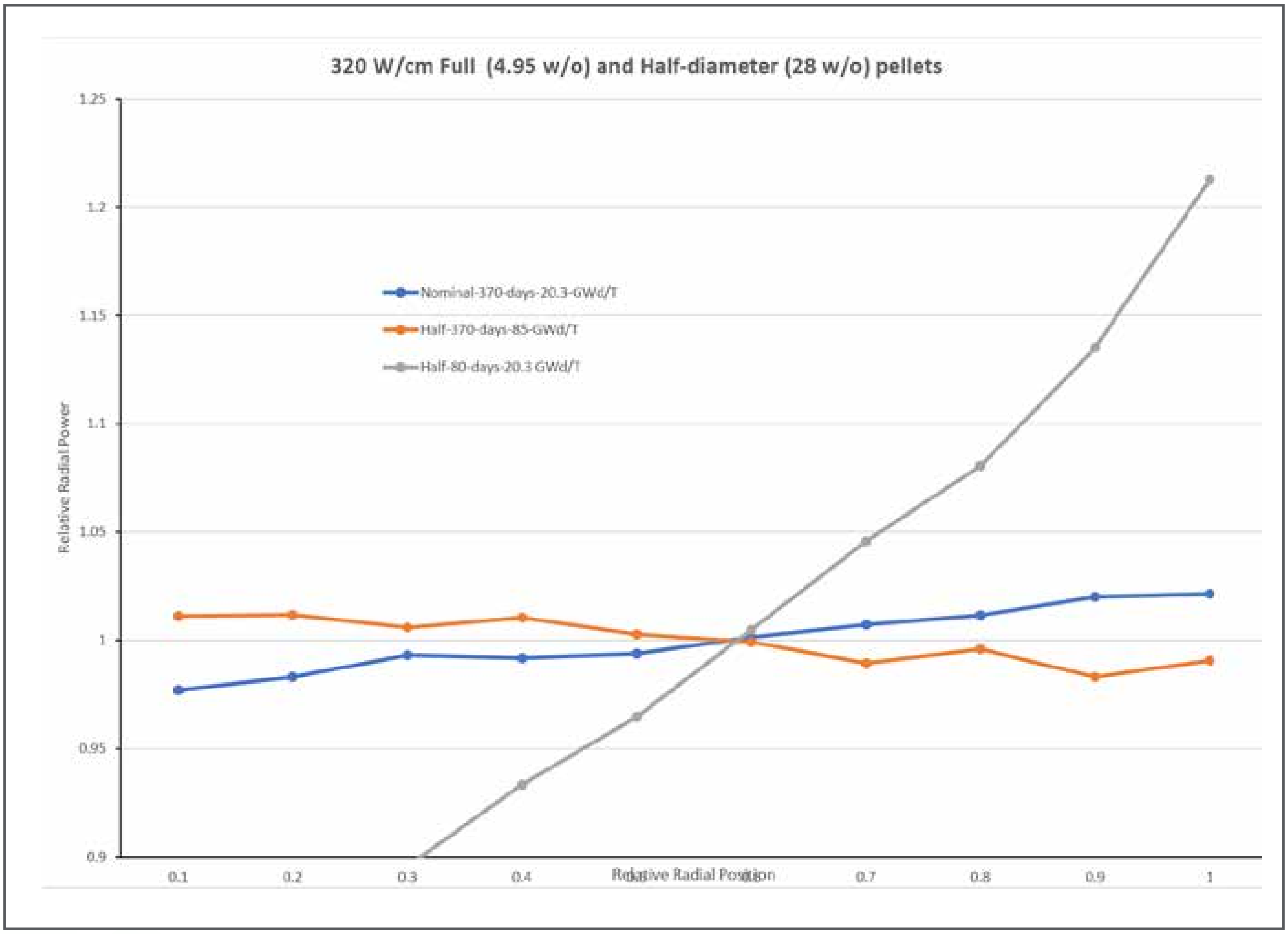


Figure 1. Serpent Model for I22 Capsule

Qualification of new fuels is a complex, and perhaps more importantly, a time-consuming undertaking. In order to obtain the data on the irradiation performance for fuels with typical light water reactor (LWR) burnups of ~50-60 GWd/t, to understand their behavior and support modeling, irradiations under "prototypical conditions" of 10-15 years are needed. Recently the Advanced Fuels Campaign (AFC) has begun exploring options to accelerate the time required to obtain the needed data. Two approaches are currently being pursued:

The simplified Serpent/MCNP models of an ATR irradiation capsule provide a capability to do quick scoping/sensitivity calculations for configurations of interest to support design of experimental irradiations.

- Reduced diameter fuel samples to increase the power density and hence increase the burnup accumulation rate at the Advanced Test Reactor (ATR), generically termed "FAST"
- MiniFuel irradiation tests at Oak Ridge National Laboratory (ORNL)

Project Description:

To support the "LWR FAST" experimental activities, simplified two-dimensional models for the irradiation capsule in the I22 position in the ATR were developed for the MCNP and Serpent Monte Carlo codes based on an MCNP model provided by Idaho National Laboratory (INL). The geometry for the capsule is shown in Figure 1. There are three (3) locations for the insertion of the test specimens. Shown in the figure is a full diameter pressurized water reactor (PWR) fuel rod that is segmented into ten (10) radial zones. A simplified surface source based on information provided by INL for the flux at the I22 position in the ATR was utilized for initial calculations of the radial power and burnup distributions as a function of irradiation time for full and reduced diameter test pellets. Fixed source calculations were performed with Serpent for nominal and half-diameter fuel pellets for a range of linear-heat-generation-rates (LHGR) typical of those for a PWR to determine the radial power and burnup distributions as a function of irradiation time. The objective of the Brookhaven National Laboratory (BNL) activities is to provide a capability to do quick scoping/sensitivity calculations for configurations of interest to support design of experimental irradiations.

Accomplishments:

Fixed source calculations were performed as described above for the irradiation of full and half-diameter UO2 fuel pellets in the I22 location in ATR for several LHGR characteristic of those for a commercial PWR. Typical relative radial power distribution are shown in Figure 2 for cases with a

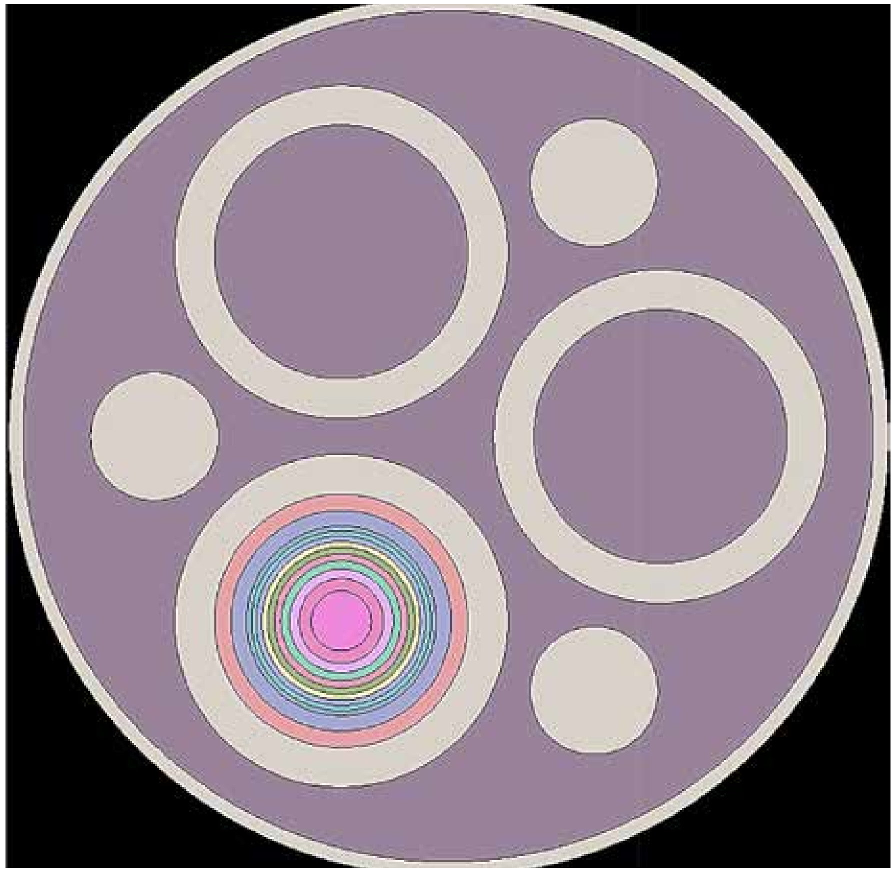


Figure 2. Relative Radial Power Distributions for Full and Half-Diameter UO_2 Fuel Pellets.

LHGR of 320 w/cc. In order to match the LHGR of a nominal fuel pellet for a LHGR of 320 w/cc with an enrichment of 4.95 w/o with a half-diameter pellet the enrichment had to be increase to 28 w/o. The three curves show the relative radial power distributions for the nominal pellet at 370-days/20.3-GWd/t and for the half-diameter pellet at 370-days/85-GWd/t and 80-days/20.3-GWd/t. These results illustrate the difficulty in connecting the results from the full and reduced diameter irradiations to provide meaningful data on fuel performance which will obviously require fuel performance modeling that incorporates these neutronic results. Analyses of "accelerated burnup" configurations such as those shown here can be used to help develop meaningful figures-of-merit (FOM) to ensure that configurations that accelerate the burnup accumulation in test fuels actually provide data that are sufficiently representative of expected fuel performance and can be used to support assessments of performance and accident behavior, and licensing activities. This will likely be a useful tool/approach to demonstrate/qualify ATF fuels, and fuels for higher burnups beyond those currently licensed.

ADVANCED REACTOR FUELS

3.1 FISSION ACCELERATED STEADY-STATE TESTING

Fission Accelerated Steady-state Testing (FAST)

Principal Investigators: Geoffrey Beausoleil
Team Members/ Collaborators: Nate Oldham, Bryon Curnutt, Kyle Gagnon, Chris Murdock and Randy Fielding

FAST is positioning INL and the DOE complex to be able to meet the modern demands of technology research and development (R&D) for nuclear fuel by reducing irradiation times for fuel up to ten times faster.

In an effort to accelerate the irradiation time for advanced reactor fuels, a revised capsule design has been analyzed and developed for the Advanced Fuels Campaign (AFC). This design incorporates a highly enriched, reduced diameter fuel pin that is doubly encapsulated by two steel capsules. This design alloys accelerated irradiations and reduced sensitivity to fabrication variances and eccentricities. The capsule designs utilize existing experiment baskets from the AFC capsules in the Advanced Test Reactor (ATR) outer A position (FAST-OA) and the ATF-1 capsules in the small I position (FAST-SI).

Project Description:

Previous advanced reactor fuels, namely fast reactor fuels, have been successfully irradiated within the Idaho National Laboratory (INL) Advanced Test Reactor (ATR). However, these tests have had challenges in that they require adjustments to the experiment design to make the power profiles more prototypic (introduction of a cadmium shroud), a tight-tolerance helium bond between the cladding and the capsule wall (a low heat flux zone for increasing the cladding temperature to prototypic temperatures), high sensitivity to fabrication variances and eccentricities (lower limit of fabrication tolerances significantly affect the outcome of the experiments), and, ultimately, very long irradiation times in order to achieve high burnup targets (10-12 years for >20 % FIMA (Fission per Initial Metal Atom)). These challenges have been addressed by the Fission Accelerated Steady-state Testing (FAST) experiment design.

Phase 1 of the FAST experiments is focused on alloy fuels for sodium fast reactors that are assembled for insertion into the ATR during the 169A cycle (November 2020). In total, there are 28 experiments being inserted:

- Control specimens that compare to historical irradiation tests will be used to understand the impact of accelerated irradiation to fuel performance. These pins are uranium with 10 wt% Zr (U-10Zr), 75% smear density, and sodium-bonded solid pins in HT9 cladding.
- Fuel Additives that contain antimony, tin, or palladium in order to mitigate lanthanide fission product attack on the cladding (fuel-cladding chemical interaction [FCCI]). These are similar to the palladium-bearing AFC-3A and AFC-3B tests but will be irradiated to higher burnups.
- Sodium-free annular fuel pins that are thermally bonded with helium. These are comparable to the helium-bonded annular fuel tests from AFC-3A and AFC-3B but will be irradiated to higher burnups.

- HT9 cladding with zirconium liners to prevent FCCI will be used with U-10Zr, sodium-bonded solid pins.

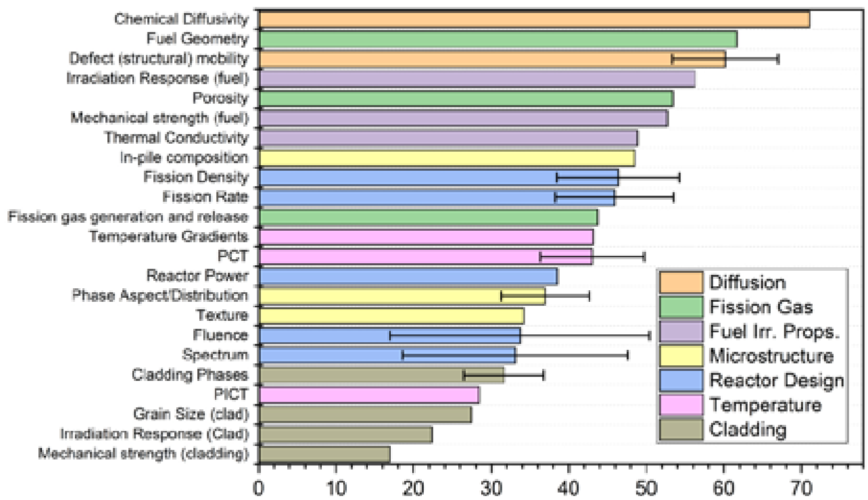
  This project will provide INL an accelerated irradiation platform that can be easily adopted to multiple fuel forms and reactor designs. The benefits will enable the Laboratory and Department of Energy (DOE) to meet industry needs for quickly investigating advanced fuel concepts and to achieve the objectives of deploying safe, reliable, and economic operation of next generation reactors.

Accomplishments:

The FAST project has completed the final fabrication and assembly of the experiments listed above and is awaiting the 169A outage for inserting them into the reactor. The project overcame numerous challenges, especially those associated with work shutdown during the initial COVID-19 response. FAST is also being incorporated into multiple research projects beyond its initial scope in AFC.

- Mockup weld processes were not successful in actual execution of the pressurized outer capsules. This was determined to be due to excessive local heating and pressurization of the head space in the capsule. The assembly team managed to respond within a week with a new weld process that reduced the time and load of the weld spark and enabled adequate closure of the weep hole. The capsules were then able to pass final inspections.
- FAST experiments were built into BISON models to begin predictive fuel performance assessments of the experiments. These models are currently being evaluated and compared to match the design analysis results (i.e., finite element thermal analysis) before turning on more extensive material models for the predictive analysis (e.g., fuel swelling, fission product distribution, or fission gas release).
- Bryon Curnutt presented the combinatorial design process of FAST at NuFuel 2019, an INL sponsored fuels conference held at the Paul Scherrer Institute in Switzerland. The presentation highlighted the methods used for designing the FAST experiments, including the scripting of experiment parameters for thermal analysis tools and neutronics analysis.
- The FAST team led a working group on accelerated qualification of advanced metal fuel for fast reactor concepts. This group included involvement from AFC and Nuclear Energy Advanced Modeling and Simulation (NEAMS) from multiple laboratories and universities. The meetings resulted in a Phenomena Identification and Ranking Table (PIRT)

Figure 1. A listing fuel behaviors and properties in order of research priority based upon the PIRT analysis done by the accelerated testing team in AFC. These priorities are being used to guide model development within BISON and post irradiation examination of FAST test specimens. Future experiment designs will be guided by those results and these property relationships.



analysis of key fuel phenomena and properties (Figure 1). The PIRT was then used to outline a modernized approach to the qualification presented by Crawford et al. [1] where stages 1-3 could be reduced from a 15-20 year process to 5-8 years (Figure 2). This PIRT and methodology for qualification is the subject of a paper submitted to *Nuclear Technology.*

- FAST is being utilized by a Laboratory Directed Research and Development (LDRD) project that is partnering with General Atomics (GA) and to irradiated uranium carbide (UC) fuel. GA is intending UC fuel compacts to be the used in their high temperature gas cooled fast reactor design (Figure 3), Energy Multiplier Module (EM^2). The LDRD is testing accelerated fuel irradiation of ceramic fuels as well as multiple fuel designs (Figure 4). UC fuel is also the subject of the Accelerated Fuel Qualification (AFQ) effort that GA has with Oak Ridge National Laboratory (ORNL) and a semi-integral irradiation test, such as with FAST, is a necessary component of qualification.

References:

[1.] Crawford, D.C., et al., *An approach to fuel development and qualification.* Journal of Nuclear Materials, 2007. **371**(1): p. 232-242.

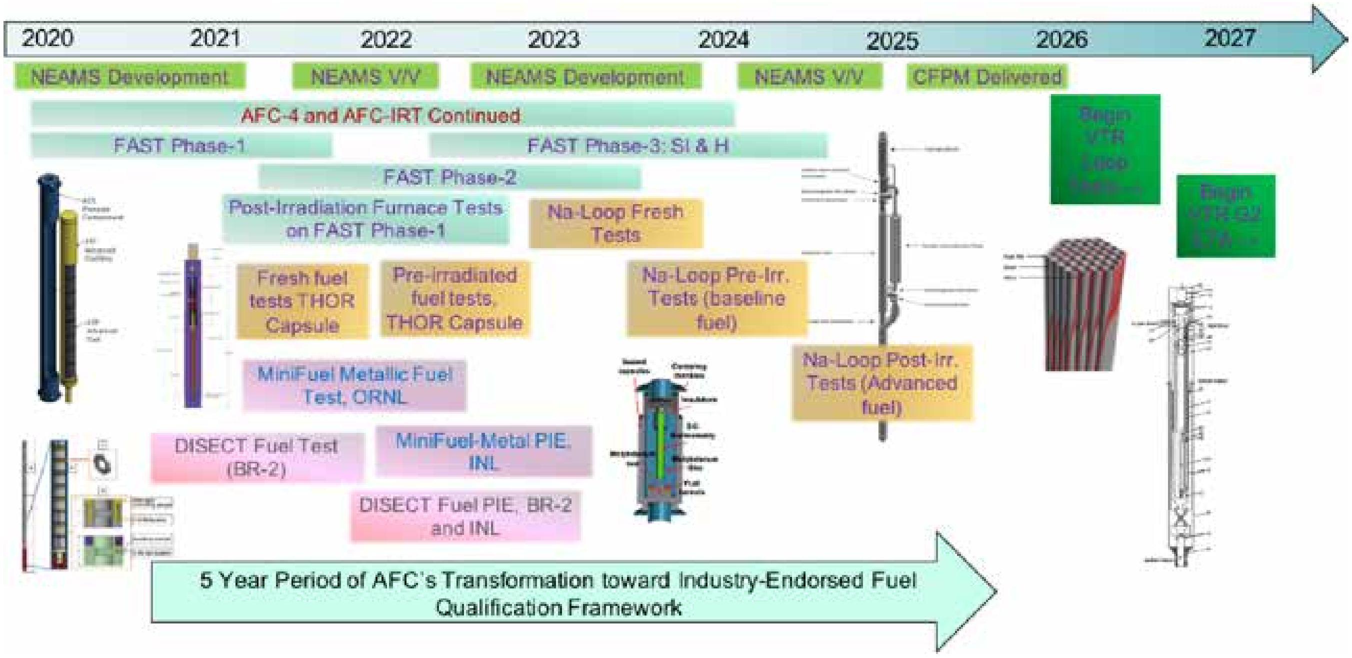


Figure 2. Integrated six-year plan for development of metal fuel for both commercial interests and the versatile test reactor (VTR) driver fuel. This plan integrated both separate effects testing in the Oakridge MiniFuel and the Nuclear Science User Facility (NSUF) Disc Irradiation for Separate Effects Testing with Control of Temperature (DISECT) experiments along with the semi-integral testing of FAST and transient conditions in TREAT.

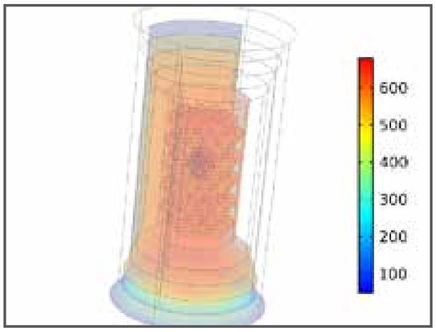


Figure 3. A thermal analysis model of uranium carbide kernel compacts to be used in a FAST experiment in support of General Atomics EM² reactor concept. The analysis was used to determine how kernel size may affect the bulk behavior of the compact as it is scaled down for FAST dimensions.

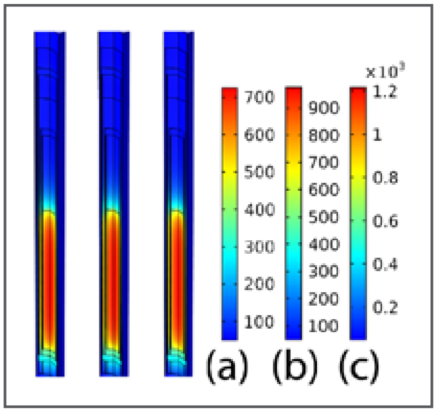


Figure 4. Three different thermal analyses of the uranium carbide FAST experiments used to determine the necessary enrichments and power generation rates needed to achieve the desired temperatures. The UC experiments will push FAST to much higher cladding and capsule temperatures than previously explored with the metal fuel, opening it up to support other very high temperature applications.

3.2 JAEA/INL ARES PROJECT AT TREAT

JAEA/INL ARES Project at TREAT

Principal Investigator: Colby Jensen and Takayuki Ozawa (JAEA)
Team Members/ Collaborators: Robert Armstrong, Cole Blakely, Luca Capriotti, Daniel Chapman, Andrew Chipman, Fidelma Di Lemma, Devin Imholte, Pavel Medvedev, Kyle Paaren, Doug Porter, Trevor Smuin, Daniel Wachs and Nicolas Woolstenhulme

The ARES joint project with INL and JAEA is underway to investigate transient performance of high burnup metallic and MOX fuels in TREAT for a world first in more than two decades.

Advanced Reactor Experiments for Sodium Fast Reactor Fuels (ARES) is a joint project between the U.S. Idaho National Laboratory (INL) and the Japanese Atomic Energy Agency (JAEA) to investigate the transient fuel performance of irradiated advanced metallic and mixed oxide (MOX) fuel designs from Experimental Breeder Reactor (EBR)-II experiment programs. Transient fuel performance of fast reactor fuels has been well-established internationally. The continued optimization of fuel designs and associated operational limits to improve performance and economics demands the continued experimental evaluation of these behaviors. These goals require the development of improved fuel performance behavioral models implemented in advanced modeling and simulation tools, which in turn require modern experiment complements with new data streams. Figure 1 shows examples of metallic and MOX fuel pins after undergoing transients. The ARES project relies on the Transient Reactor Test (TREAT) facility and the development of associated in-pile testing infrastructure to provide SFR boundary conditions to accomplish its goals.

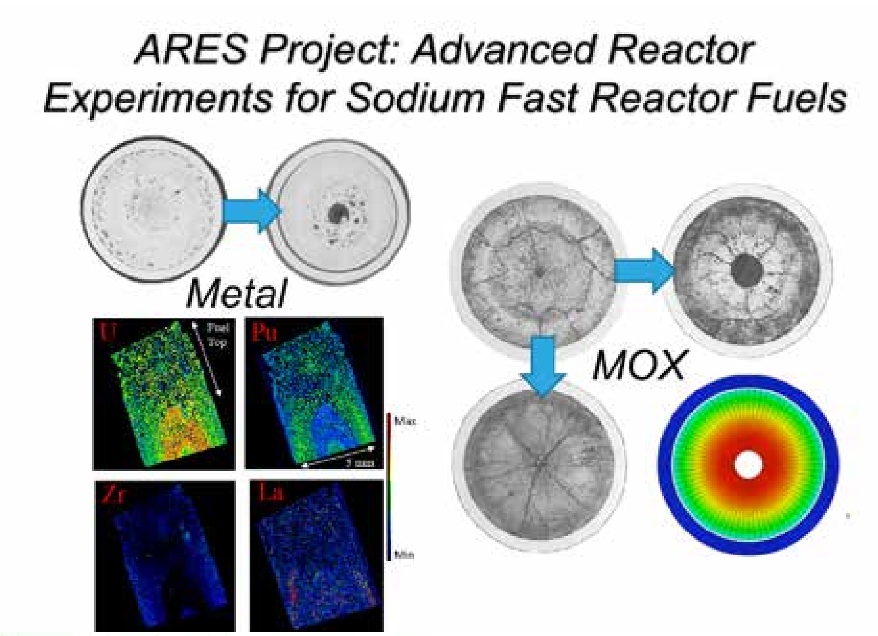


Figure 1. Graphical representation of the ARES project for metallic and MOX fuel experiments in TREAT with U.S. DOE and JAEA.

Project Description

The ARES project will investigate the transient performance of high value irradiated metallic and MOX fuels as a collaboration between INL and JAEA to support improved performance of advanced reactors. The objectives are to investigate fuel failure modes in high burnup metallic and MOX fuels while developing and validating transient fuel performance models and establishing testing infrastructure at the TREAT facility. Figure 2 provides an overview of project timelines and tasks covering fiscal year (FY)20-FY23. The ARES project is comprised of five primary components in this first phase.

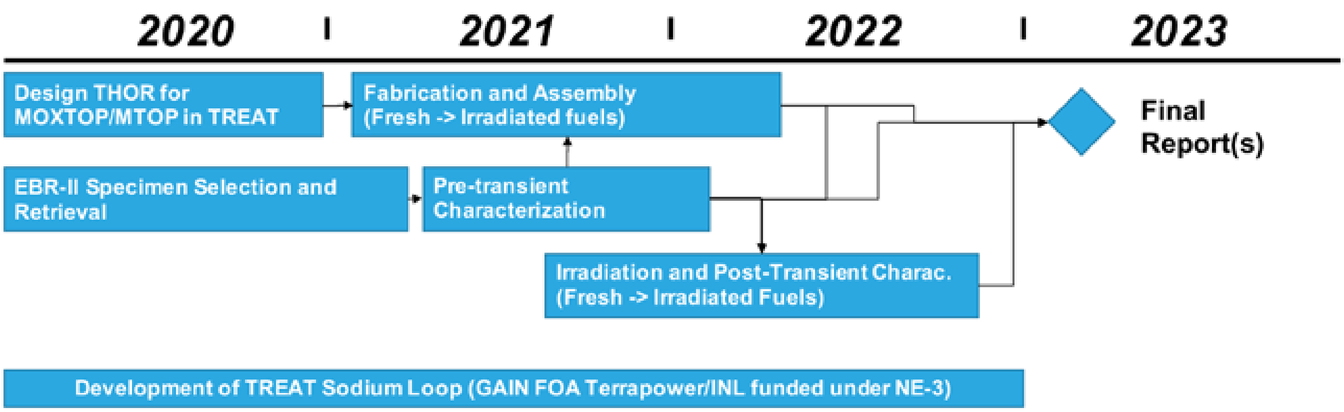


Figure 2. Overview of ARES project schedule leading to transient experiments in TREAT in FY22.

1. Design of an in-pile SFR testing capsule and associated hot cell infrastructure to remotely load irradiated fuel specimens using the best modeling and simulation capabilities.
2. Fresh fuel commissioning tests to qualify the TREAT testing platform for providing required nuclear heating and thermomechanical boundary conditions and measuring the associated fuel response.
3. Four integral experiments will be performed in TREAT including two irradiated metallic fuel specimens (called M-TOP/M-LOF series) and two MOX fuel specimens (called MOXTOP series).
4. Pre- and Post-Transient Examinations will be performed on the EBR-II-irradiated metallic and MOX fuel test specimens.
5. Data synthesis and detailed experimental evaluation will be performed using BISON and JAEA fuel performance codes.

Planned experiments will be performed in the Temperature Heat sink Overpower Response (THOR) module in the Minimal Activation Retrievable Capsule Holder (MARCH) system in TREAT. The THOR test device is comprised of a capsule to house a fully intact EBR-II metallic or MOX fuel pin. The design includes a metal heat sink surrounding the specimen with a thin sodium annulus to provide excellent thermal conductance from the specimen to the heat sink. The design process relies heavily on advanced modeling tools to guide experiment design and objectives. Fresh fuel experiments on U-10Zr will commission the device capability in the TREAT Facility in FY21 to FY22. These tests will provide valuable insight on performance of advanced in-situ instrumentation including optical

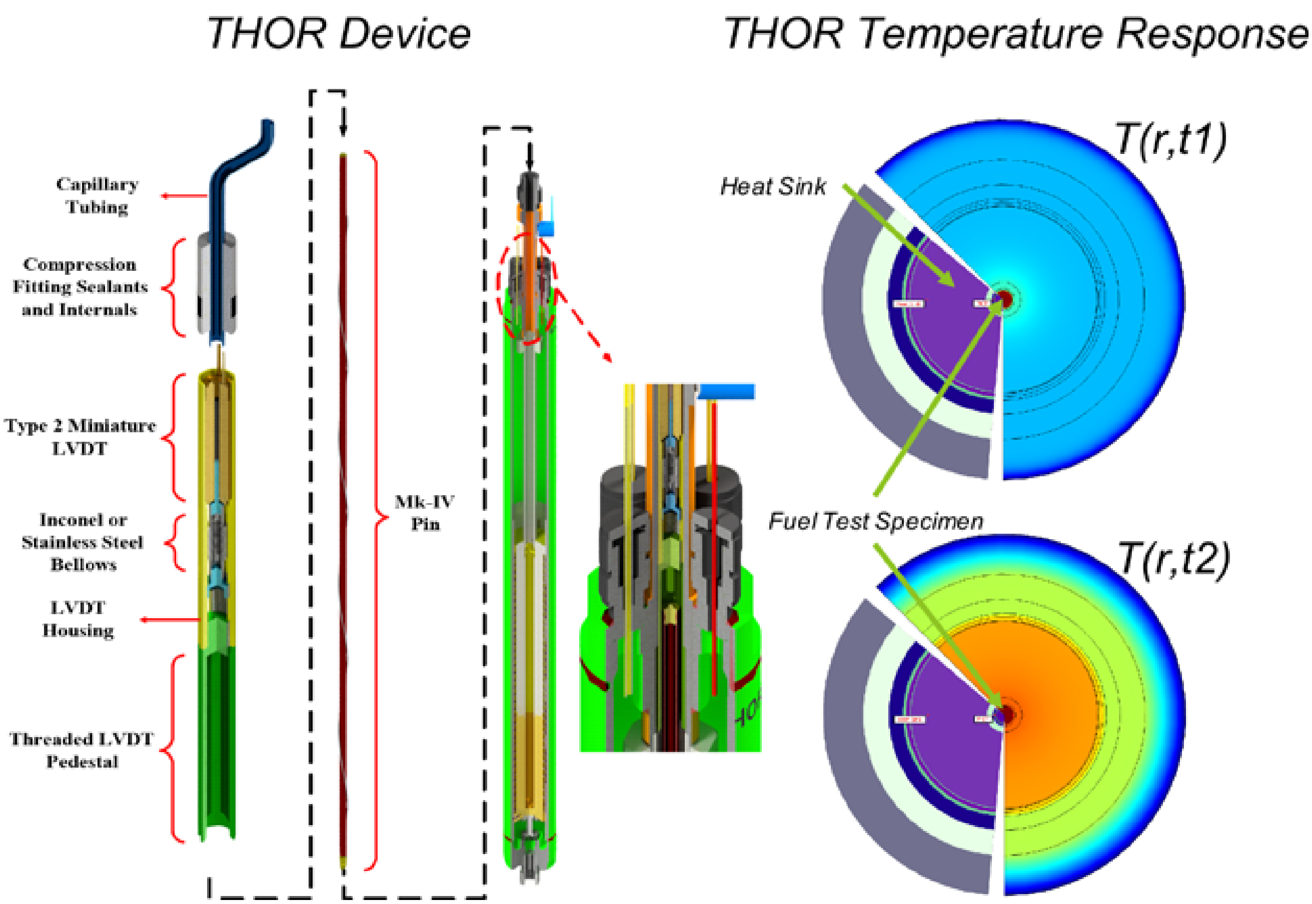


Figure 3. Schematic overview of the THOR device for TREAT with examples of cross-section illustrating temperature response in an experiment.

fiber sensors, infrared pyrometers, Linear Voltage Differential Transformers (LVDTs), and the TREAT Fuel Motion Monitoring System (AKA hodoscope). Pre-transient characterization of irradiated specimens will also begin in FY21. Hot-cell assembly and irradiation of irradiated specimens will occur in FY22 followed by post-transient examinations. The results of these tests will be synthesized in detail and compared to fuel performance simulations pre and post experiment.

These experiments are poised to establish crucial infrastructure at TREAT and associated hot cells to address existing data gaps and extend the existing database while paving the way to future experiments. The outcomes will directly serve needs of the U.S. and Japanese SFR research and commercial communities.

Accomplishments

As the first year of a multiyear project, the project kicked off by completing detailed work scope and creation of a CRADA between INL and JAEA through FY23. Since then, several activities have been underway that have included the conceptual through final design of the THOR test device shown in Figure 3. The design and analysis supporting the design have shown the unique advantages and limitations of the relatively simplistic THOR approach to passively produce thermal boundary conditions. The analysis clearly demonstrates its complementary function with the sodium loop currently under joint development by INL and Terrapower. A wide range of relevant time vs. temperature conditions are possible using the THOR device encompassing the conditions of interest for MOX and metallic fuels. Some design challenges that have been overcome include incorporation of full-length, intact EBR-II pins, incorporating a range of advanced instrumentation to ensure data objectives are met, and assembly considerations required for remote glove box and hot-cell facilities. As part of this process, a fresh fuel commissioning experiment series has been developed to measure/calibrate specimen power in TREAT and demonstrate and qualify the system performance for use with high value irradiated specimens. A detailed conceptual design report was provided to JAEA as a deliverable in March 2020; the preliminary design for the THOR device was held in June 2020; the final design review for the device is set to begin in September 2020.

In parallel to device design and modeling activities, the experiment design effort has focused on development of specimen configuration details. Working with the legacy database and JAEA, irradiated metallic and MOX fuel specimens have been located in an INL fuel storage facility to support the testing objectives. The pins for the MOX tests have been identified from EBR-II experiment X462A (or also called SPA-2B). The pins were irradiated to nearly 130 GWd/t in PNC1520 cladding with as-built annular fuel configuration. For the metallic fuel experiment, pins allocated for the planned but never executed M8 test have been located with additional sister pins that will be used in the M-TOP/M-LOF experiments. These pins were irradiated to nearly 13 at% burnup in EBR-II. Sister pins were also tested in overpower tests in EBR-II as well as in the historic Whole Pin Furnace for LOF simulation. The as-built documentation and irradiation histories for all pins have been recovered and fuel performance models using BISON and JAEA code are now nearing completion. Preparations are underway to transfer both sets of pins to Hot Fuel Examination Facility (HFEF) for pre-transient characterization and eventual experiment assembly.

While comprised of important goals and many challenges, the project has made significant progress in its first year to complete the final design of a heat sink capsule to support broad Sodium Fast Reactor (SFR) studies and beyond, but also identification, recovery, and models of legacy advanced test specimens. The first modern experiments on advanced sodium reactor fuels will be performed in FY21. The THOR device has already been identified by multiple other programs a key capability to meet their needs. The transient and characterization data will support development of advanced modeling capabilities (working with Nuclear Energy Advanced Modeling and Simulation (NEAMS) for BISON code in U.S.) and aid U.S. and Japanese advanced reactor industries to provide safe and economic nuclear energy.

3.3 SCANNING ELECTRON MICROSOPY EXAMINATIONS

Scanning Electron Microscopy Examinations of Advanced Metallic Fuel Forms

Principal Investigators: Luca Capriotti
Team Members/ Collaborators: Tammy L. Trowbridge, Fidelma Di Lemma and Jan-Fong Jue

Scanning electron microscopy on several innovative fuel designs have revealed insights on their microstructure and have shown good performance in pile at low burnup.

In sodium fast neutron spectrum nuclear reactors, metallic uranium-based alloys have often been chosen because of their high fissile density, high thermal conductivity, and reactor safety benefits. In order to address increasingly high burnup / performance demands, a number of innovative fuel designs are under investigation within the Advanced Fuels Campaign (AFC). As part of this development, candidate fuel compositions and forms are irradiated in a cadmium-shrouded positions at the INL's Advanced Test Reactor (ATR), and they are subsequently examined at the Material Fuel and Complex (MFC) facilities.

Recent irradiation experiments have explored new alloys and geometric forms beyond what has historically been irradiated (U-10Zr, 75% smeared density, sodium bonded fuel) to overcome two primary limiting performance factors: high swelling rate at higher burnup and fuel cladding chemical interaction (FCCI) from lanthanides fission products.

Project Description:

The innovative fuels design, explored in these experiments, fall into various categories: alternative alloying metals, additives, sodium bond removal, and lower smear densities. For alternative alloys, Zr has been substituted with elements that stabilize the cubic phase of U such as Mo to prevent constituent redistribution. Additives, such as Pd have been suggested to bind lanthanide fission products, effectively reducing FCCI. Sodium bonding is considered undesirable at the back end of the fuel cycle where sodium bonded fuel must be treated to remove sodium prior to geological disposition. Helium bonded metallic fuel of several alloys has been irradiated to better understand the implications of removing sodium bonding. Finally, low smear density fuel is necessary to enable high (30 at.%) burnup and both solid and annular low smear density fuels have been examined.

Postirradiation examination (PIE) and microscopy exams such as scanning electron microscopy (SEM) have been completed on several advanced fuel forms from the AFC-3 series experiments to assess the performance in pile, characterize the microstructure evolution under irradiation and to compare the overall behavior with metallic fuel literature. Metallic fuel alloys investigated are U-10Zr annular fuel and helium bonded, Pd bearing metallic fuel in solid and annular form.

Accomplishments:

Recently performed SEM characterization on two annular metallic fuel specimens, U-10Zr irradiated to 4.3 %FIMA (Fission per Initial Metal Atom) and U-13Zr-4Pd irradiated to 2.7 %FIMA, have provided first

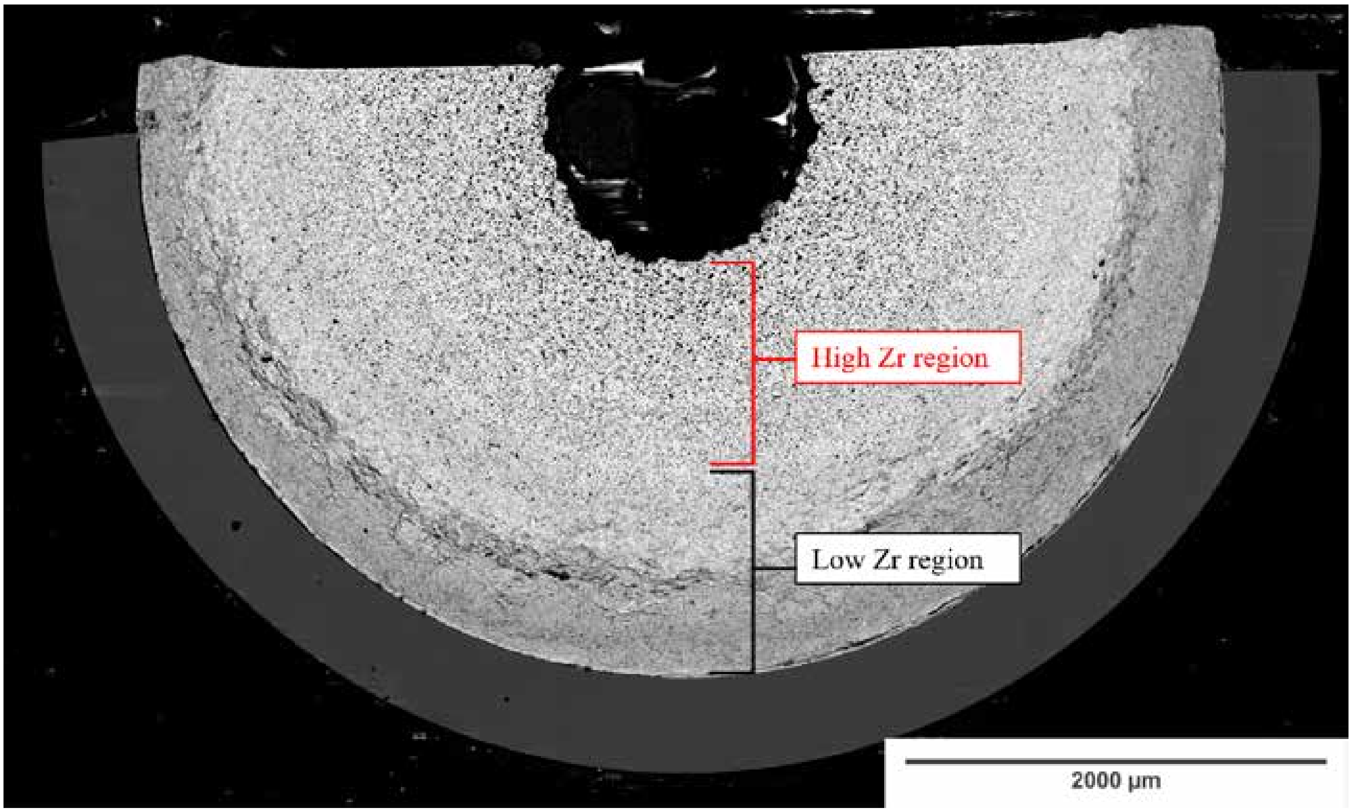


Figure 1. Backscatter electron montage image of half fuel cross-section of U-10Zr annular fuel irradiated as part of AFC-3D experiment to a burnup of 4.3 %FIMA. In the image, different microstructures across the radius are visible and Zr-redistribution is also indicated as "high Zr region" and "low Zr region."

of a kind and important insight on the performance of the previously described innovative fuel design.

The microstructure of U-10Zr annular fuel is visible in Figure 1 and presents different degree of porosity along the radius. The inner band shows round and large porosity and a predominant single fuel phase that during reactor operation was likely gamma-(U,Zr) phase (temperature higher than 650 °C) [1]. From energy dispersive X-ray spectroscopy (EDS) analyses, this central part has a higher concentration of Zr compare to the as-fabricated composition which means that Zr/U redistribution has occurred as expected, in line with the metallic fuel literature. In the outer region the Zr content is lower and the porosity is somewhat smaller and elongated. Finally, there is no evidence of wastage inside the cladding as part of the FCCI region.

Overall, this U-10Zr annular fuel has shown good performance in line with the well-known U-10Zr, sodium

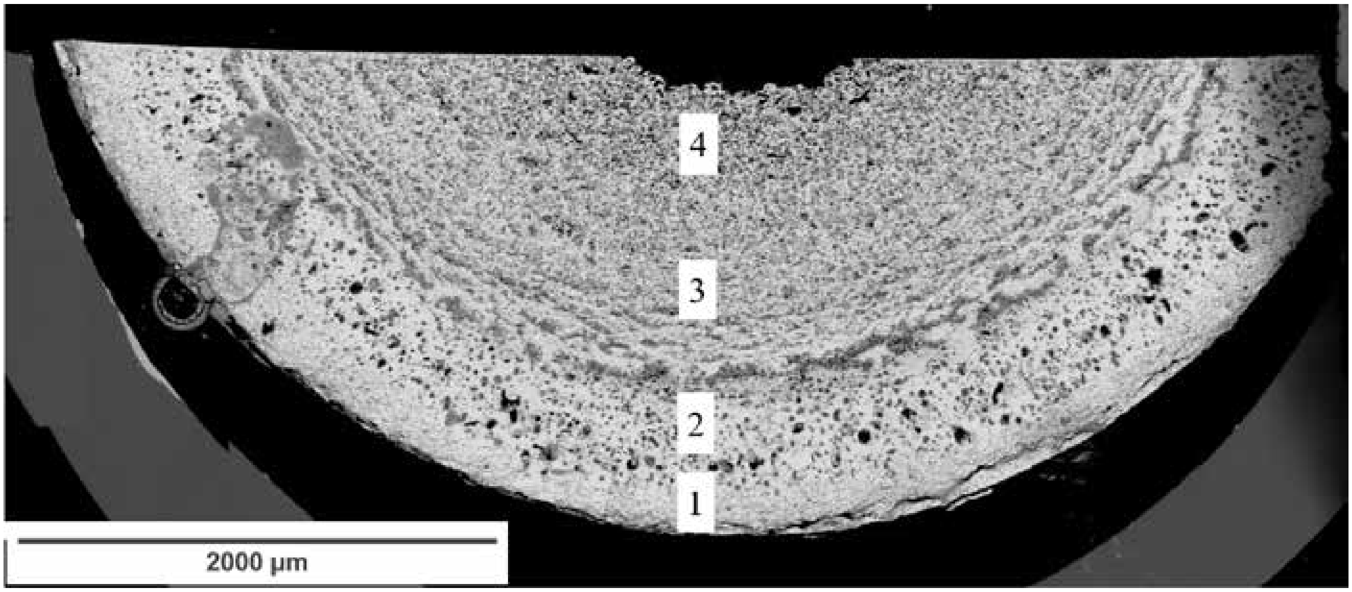


Figure 2. Backscatter electron montage image of half fuel cross-section of U-13Zr-4Pd annular fuel irradiated as part of AFC-3D experiment to a burnup of 2.7 %FIMA. In the image, different microstructures are evident and identified in 4 main regions (from 1 to 4). Most of the cladding is not visible and detached from a closed contact with the fuel during sample preparation.

bonded fuel, thanks to an on-going optimization of the design and fabrication [1].

As already mentioned before, palladium has been proposed as an additive to metallic fuel to help prevent lanthanide-initiated FCCI in U Zr alloys at high burnup [2,3]. During fabrication, Zr and Pd form a stable intermetallic compound, and Pd was expected to release Zr and form stable intermetallic compounds with lanthanides. This concept has been tested earlier [4] discovering issues attributed to Zr Pd intermetallics that removed Zr from the U matrix, and leading to increase interaction in between U and the cladding. To help mitigate this problem in a later AFC-3 test (reported here) the fuel composition was changed with addition of

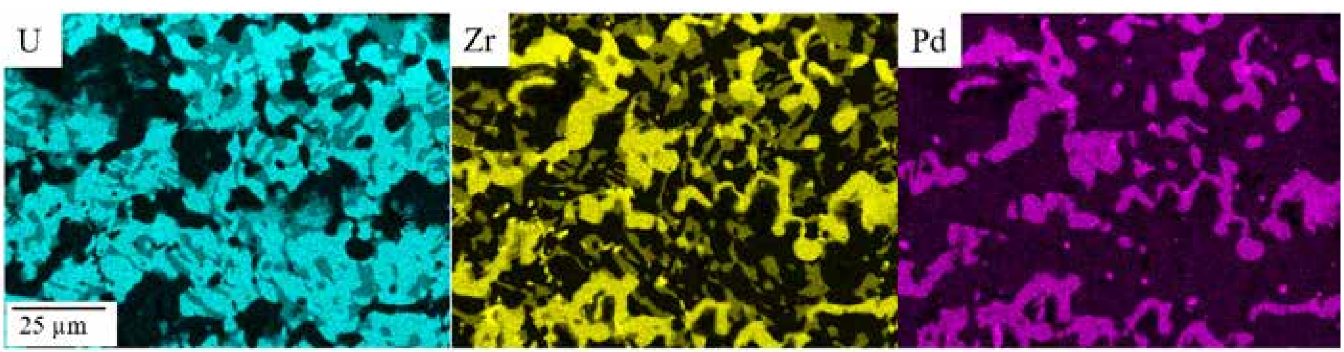


Figure 3. Backscatter electron images showing in detail the different regions from Figure 2. In the periphery of the fuel (region 1-2) Pd is found mainly associated with rare earth, while in the central regions (3-4) Pd-Zr precipitates are the majority.

more Zr (13% not 10%) to the Pd bearing pins. The U-13Zr-4Pd annular fuel microstructure is visible in Figure 2 (unfortunately the cladding detached from a closed contact with the fuel during sample preparation). In general, it seems not very dissimilar compare to the one from U-10Zr annular and the fuel radius can be divided in 4 regions as indicated in Figure 2. A more in detail description of the different regions is shown in Figure 3 from which emerge that Pd is more likely to bond with rare earths at the periphery of the fuel (region 1-2), instead in the central regions (3-4) the majority of Pd is alloyed with Zr, forming an intermetallics very close to $PdZr_2$. Figure 4 shows also an EDS maps of U, Zr and Pd from region 3 in which is clear the association Zr-Pd. Finally, regarding U and Zr in the matrix, there is everywhere in the fuel the presence of a U-rich phase and a second phase U-Zr with Zr variable from 10wt% (region 1-2) to 35wt% (region 3-4).

The performance of this optimized Pd-bearing fuel alloys seems to have improved and do not show any obvious performance difference between U Zr pins with the same geometry at this burnup. The difference in behavior of the intermetallic formations between center and periphery and the efficiency of Pd as additive to mitigate FCCI needs more deep understanding and further verification at higher burnup.

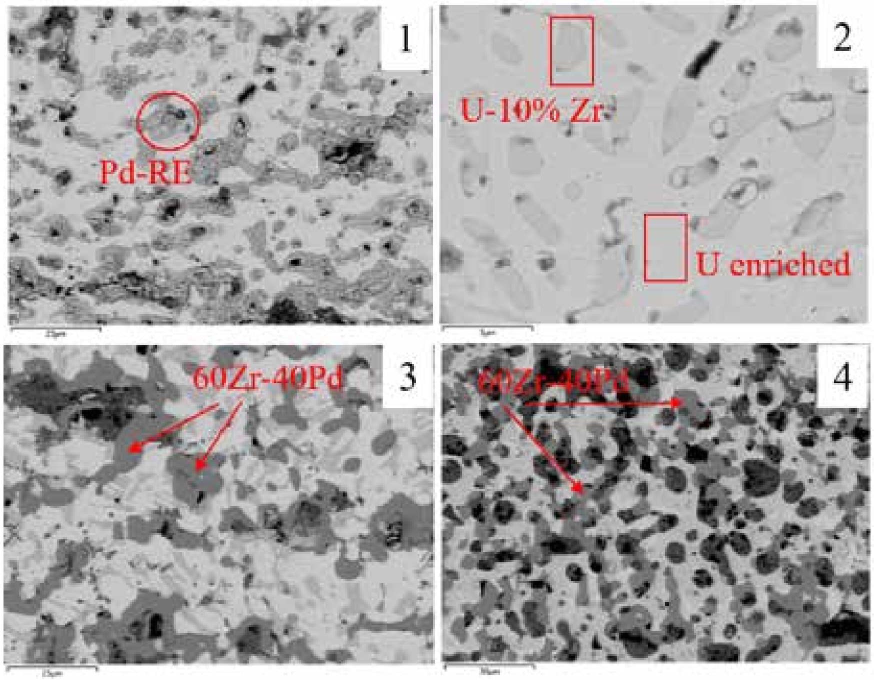


Figure 4. Energy dispersive X-ray spectroscopy (EDS) maps of region 3 from Figure 2 highlighting the distribution of Zr, Pd and U.

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3.4 STATUS AND PROJECTIONS

Status and Projections for Conclusion of the Long-Running AFC Series Irradiations

Principal Investigator: Nicolas Woolstenhulme, INL
Team Members/ Collaborators: Randy Fielding, Bryon Curnutt, Colby Jensen, Chris Murdock, Doug Porter, Dan Chapman and Brian Gross

The long-running AFC irradiations have made numerous advances in the understanding of fuel performance and thus paved the way for a new era of future advanced fuel developments.

The AFC series of irradiation tests have long been underway at the Advanced Test Reactor (ATR). This test series has addressed numerous research and development needs for many of the key benefit areas of advanced fuels, especially in fast reactor application, including actinide transmutation, fuel utilization through ultra-high burnup fuels, and direct disposal once-through metallic fuel systems. The AFC test series began just after the turn of the millennium and have proceeded in a continuous program for nearly 20 years to the present day. In 2020 unexpected shortfalls in funding, combined with the advent of new approaches to accelerated irradiation testing and final preparations for a beryllium replacement long outage at ATR, have set the backdrop for retiring the workhorse AFC series irradiations. This summary describes the background, current state, and projections for completion of the remaining AFC irradiation specimens.

Project Description

The AFC tests approximate fast reactor test conditions in ATR by placing fuel rodlets in capsules with gas gaps to elevate specimen temperature while surrounding them with cadmium to filter thermal neutrons. The AFC irradiations have been divided into 4 major campaigns to date (AFC-1 through -4) with several sub-campaigns (e.g. AFC-4A, -4B, etc.). Some rodlets can take several years to achieve burnup targets. As a result, sub-campaigns typically represent a new batch of rodlets shuffled into the irradiation positions during reactor outages. New sub-campaigns often corresponded with fabrication developments, sharpened research focus areas resulting from Post Irradiation Exams (PIE), and other refinements based on input from collaborators partners.

The AFC-1 and -2 irradiations are now complete and successfully addressed research needs for several fuel compositions including both metallic and ceramic fuels types showing the relative insensitivity of fuel performance when adding actinides. The AFC-3 and -4 campaigns have focused on advanced metallic fuels focusing largely on two objectives including fuel designs able to 1) perform well without fuel-to-cladding bond sodium (thus enabling direct disposal fuel cycles) and 2) mitigate the life-limiting phenomena associated with fuel cladding chemical interaction (FCCI). The former of these objectives has focused development on the use of annular fuel geometries in close

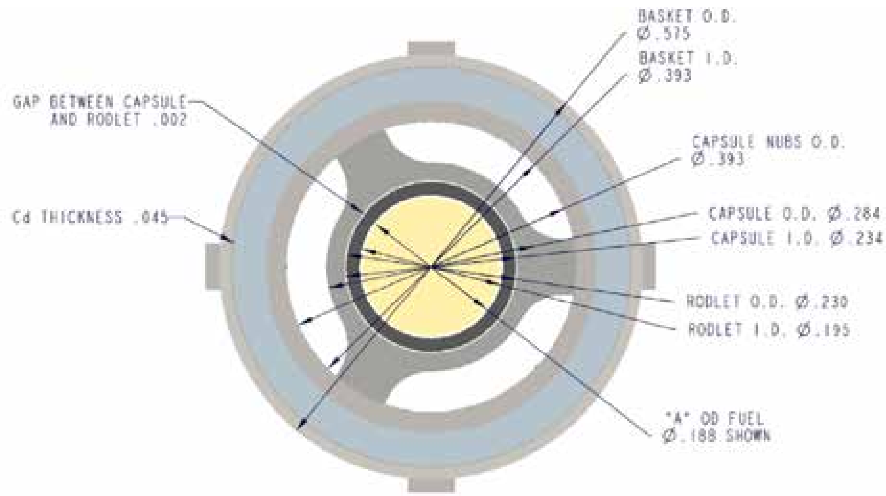


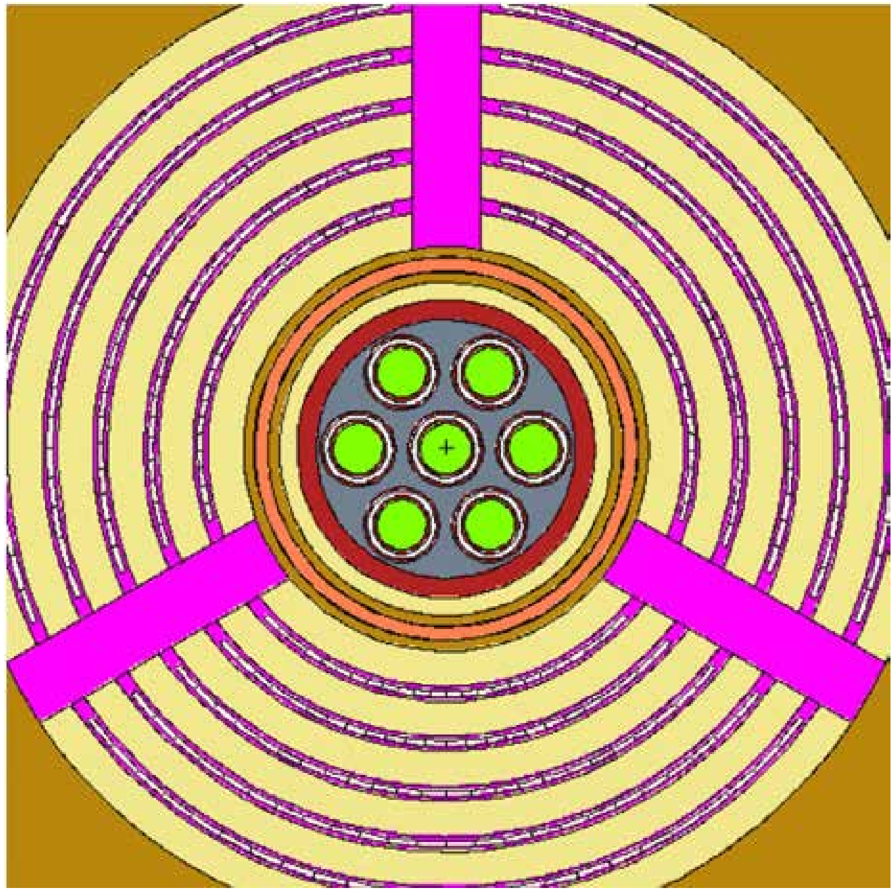
Figure 1. Typical Cross Section of AFC-Series Irradiation Tests.

contact cladding where the latter objective has focused developments both on the use of FCCI-mitigating cladding internal liners and the use of fuel alloy additives which tend to form stable intermetallic compounds with lanthanides. Most of the AFC-3 specimens, and some of the AFC-4 specimens, have already competed their planned irradiation. Presently 16 rodlets are undergoing their final irradiation cycles. These rodlets are from the AFC-3F, -4B, -4C, and -4D sub-campaigns along with rodlets from the first phase of the Integral Recycling Test (AFC-IRT1); a collaborative effort undertaken with the Korean Atomic Energy Research Institute to evaluate the effects of using pyrometallurgically recycled light water reactor fuel in metallic fuels.

Accomplishments

During fiscal year 2020 the AFC irradiation team oversaw the start of three normal-length (~60 day) irradiation cycles for continued irradiation of the experiments noted above. The AFC-3F, -4C, and -IRT1 tests are expected to complete their current irradiation in early autumn of 2020 while rodlets from the AFC-4B and -4D will continue one more irradiation cycle ending in winter of 2021 just prior to ATR's shut down for a 1-year beryllium replacement outage. These rodlets will be allowed to cool and shipped for PIE in 2021. Each of these rodlets is expected to achieve a different final burnup level in accordance with its objectives. An impressive final peak burnup slightly exceeding 20 at% is expected in two of the AFC-4B rodlets.

Figure 2. Neutronic Model Rendering of the ARCTIC Concept.



The AFC irradiation team also performed design and modeling to finalize a test matrix and began tooling up for fabrication of the previously planned AFC-4F and -G tests. This matrix included irradiation of U-Zr and U-Pu-Zr alloys with promising FCCI-mitigating liners, strategic inclusion of rare earth elements to help accelerate FCCI effects (a form of accelerated testing to help hasten liner type selection), and the use of annular extruded and slotted fuel designs to help enable direct disposal sodium-free designs.

These efforts were designed to be synergistic with the first Fission Accelerated Steady-state Test (FAST-1) irradiation. FAST rodlets use an innovative approach where radial dimension reduction and enrichment increase are used to manage thermal conditions with boosted fission rates. Accordingly, the AFC-4F and -4G were designed to focus on geometry and scale phenomena at representative radial dimensions for comparable designs. Unfortunately, congressional approval of the overall program budget in winter of 2020 brought an unexpected halt to preparation of the AFC-4F and -G tests, but the test matrix itself remains an intellectually sound product expected to be useful in future developments.

The phase 1 AFC-IRT1 project is planned to end in December of 2020 with the prospect of a phase 2 project still uncertain. Design enrichment calculations and fabrication efforts were initiated to construct a handful of rodlets to be used in a potential phase 2 irradiation (-IRT2). These rodlets will be notable in using source material from recycled light water reactor fuel rather than the simulated compositions used in -IRT1. The -IRT2 rodlets are expected to be fabricated prior to the end of calendar year 2020 and placed in storage with "ready to irradiate" status while awaiting potential commencement of a potential phase 2 project.

The successful fabrication and imminent irradiation of the first FAST experiments marks the beginning of a new era for small specimen capsule irradiations in ATR. As such, it is expected that the AFC series will be largely retired (with perhaps the exception of continued irradiation of the -IRT2 specimens). The diminutive size of FAST rodlets (~2 cm fuel), however, will be inadequate to address future data needs such as full-scale effects of fabrication process and safety performance testing in the Transient Reactor Test facility. A small project funded by laboratory directed research and development was undertaken in 2020 to investigate methods for large scale boosted fast flux irradiations in ATR. This concept, referred to as the Advanced Reactor Concepts In Thermal-spectrum Capability (ARCTIC), uses commercially available plate type fuel assemblies in a flux trap, in concert with cadmium neutron filtering, to achieve an predicted fast-to-thermal neutron ratio of 100:1. Multiple rods up to full length with representative linear heating rates can be accommodated in a novel high conductivity capsule design while providing opportunities for in-reactor instrumentation. As the program looks forward to a revitalization of resources in order to support advanced reactor metallic fuel development, it is expected that the legacy of the AFC series irradiations will live on in a new generation of test series within FAST, ARCTIC, and beyond.

3.5 HIGH DOSE MATERIALS TESTING

High Dose Materials Testing for Fast Reactors

Principal Investigator: Benjamin Eftink
Team Members/ Collaborators: Tarik Saleh (LANL), David Hoelzer (ORNL), Cheng Xu (TerraPower) and Stuart Maloy (LANL)

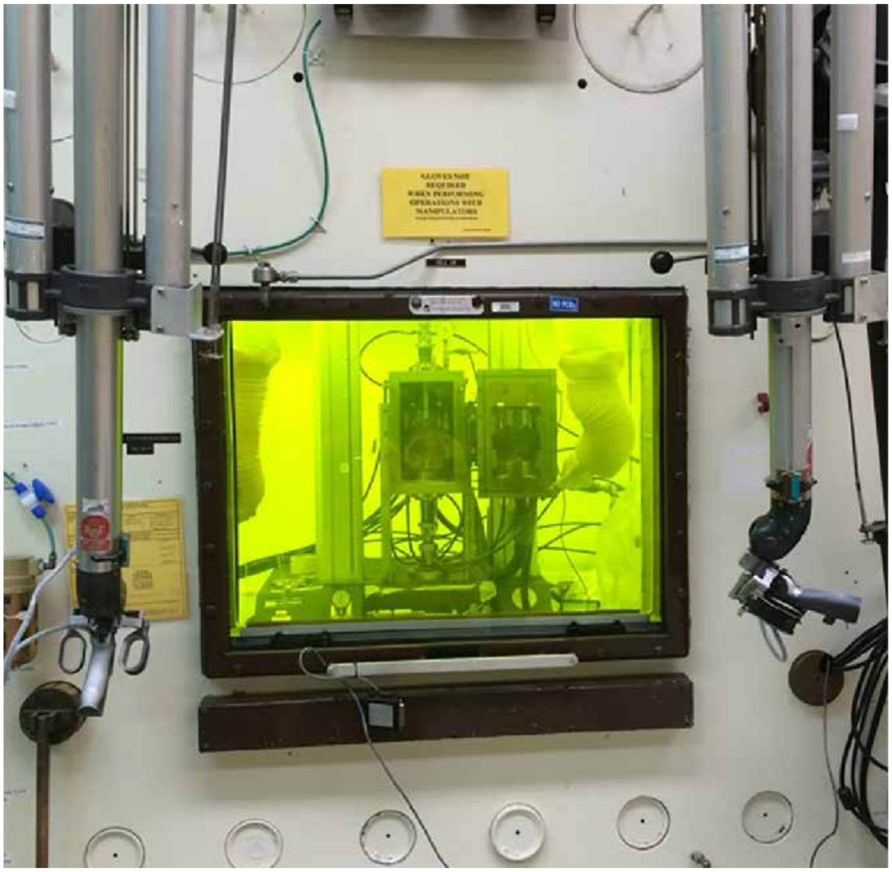


Figure 1. Image of hot cell at Los Alamos National Laboratory (LANL) in Chemistry and Metallurgy Research (CMR) Wing 9, where mechanical testing occurred.

The main goal of the Nuclear Technology Research and Development (NTRD) program is to maximize nuclear fuel utilization while minimizing high level nuclear waste. In both the modified open and closed fuel cycle options these goals are accomplished through irradiating the nuclear fuels to very high burnup (>40%) in a fast neutron flux. In order to develop fuels to accomplish this grand challenge, core materials (cladding and ducts) must be fabricated and tested to be able to withstand this challenging environment. This environment includes: 1) radiation to displacement doses greater than 400 displacements per ataom (dpa) that promote low temperature embrittlement, radiation induced segregation, high temperature helium embrittlement, swelling and accelerated creep; 2) corrosion with the coolant, and 3) chemical interaction with the fuel (FCCI). Thus, data is needed on the effects of irradiation on materials to doses greater than 400 dpa.

Project Description:

Ferritic/martensitic (FM) steels and nanostructured ferritic alloys (NFA) are being developed for next generation high dose reactor applications. Applications such as transmutation of long lived isotopes in used

Understanding mechanical properties of structural materials after high irradiation dose is necessary for designing advanced reactors.

fuels require cladding materials that can withstand high irradiation doses, at intermediate to elevated temperatures and in corrosive environments. Cladding materials for Generation IV fast reactors, such as the sodium fast reactor, will be subjected to irradiation damage levels in the hundreds of dpa and temperatures of 400-600 °C, which is significantly higher than the requirements for current generation light water reactors. Lower doses are expected in sodium fast reactors at lower temperature (e.g., 350-400 °C) regions, but strong hardening is observed at these temperatures. Thus low temperature irradiation data is important to characterize mechanical properties.

FM steels such as HT-9 have been used historically for fast reactor fuel cladding and in-core structural materials including in the Experimental Breeder Reactor (EBR-II) fast reactor and the Fast Flux Test Facility (FFTF). HT-9 steel, is a Fe-12Cr steel with additions of Mo, Mn, Ni, V and W. With a BCC crystal structure and a high density of sinks for defects including dislocations, martensitic lath boundaries, and precipitates, HT-9 has a low steady state swelling rate and long incubation period. HT-9 also has excellent corrosion resistance in sodium. Modern NFA's including the oxide dispersion strengthened (ODS) steel, 14YWT, improve upon the traditional ferritic/martensitic (F/M) steels with better high temperature mechanical properties and swelling resistance. The uniform dispersion of nano-scale oxide particles act as barriers to dislocations as well as recombination sites for irradiation induced vacancies and interstitials. This project provides mechanical testing on high dose neutron irradiated materials, comparing traditional materials such as HT-9 to advanced alloys including 14YWT.

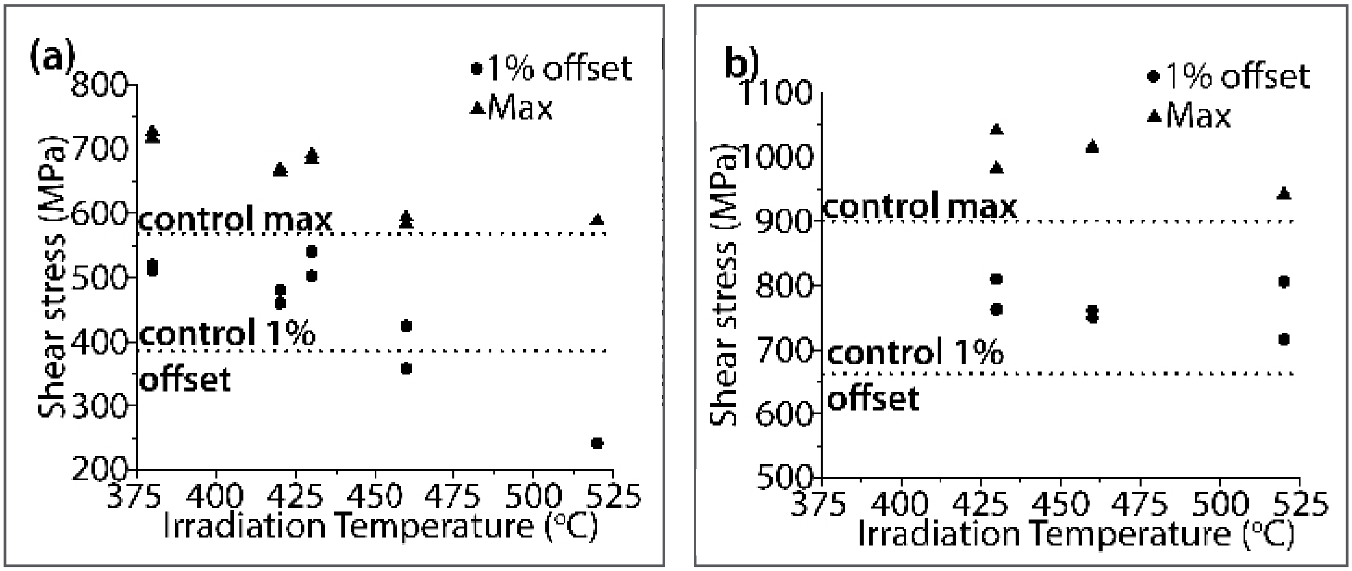


Figure 2. Comparison of 1% offset effective shear stress and maximum effective shear stress at doses between 15 dpa and 20 dpa for (a) HT-9 and (b) 14YWT as a function of irradiation temperature. Dotted lines mark the average, maximum, and 1% offset shear strain for the unirradiated control samples tested at room temperature. The average control maximum shear stress was 566 ± 17, and the 1% offset was 384 ± 37 MPa for HT-9. The average control maximum shear stress was 903 ± 14 and the 1% offset was 661 ± 25 MPa for 14YWT.

Accomplishments:
Shear punch mechanical testing was performed on neutron irradiated HT-9 and 14YWT alloys comparing the traditional FM steel to the advanced ODS NFA. The samples were irradiated at various temperatures between 380 and 520 °C. In HT-9, by increasing the irradiation temperature, the 1 % offset shear stress and maximum shear stress decreased. The advanced 14YWT alloy was less sensitive to irradiation temperature with respect to irradiation hardening. However, it still exhibited irradiation hardening at each temperature.

CAPABILITY DEVELOPMENT

4.1 FUEL ROD RE-FABRICATION AND INSTRUMENTATION

Fuel Rod Re-fabrication and Instrumentation

Principal Investigator: Jason Schulthess
Team Members/ Collaborators: Dan Wachs, Colby Jensen, Joe Palmer, Spencer Parker, Evans Chambers, Kim Davies, Gene Matranga, Cad Christensen, Jordan Argyle, Mark Cole, Collin Knight and Todd Pavey

The ability to extract fuel from the reactor, re-fabricate, and install instrumentation at the critical testing points is an essential capability that does not exist today.

Follow-on irradiation testing of previously irradiated fuel rods enables highly instrumented experiments to be performed. This testing allows researchers the ability to assess the state of the fuel at any point in its life cycle. Follow-on irradiation testing is currently targeted to be performed using the Transient Test Reactor (TREAT) and Advanced Test Reactor (ATR) at Idaho National Laboratory (INL). The ability to process preirradiated fuel rods of any length to achieve the desired form factor and include instrumentations providing online measurements is a crucial capability to support testing of Accident Tolerant Fuel (ATF) fuels.

Project Description

Performing irradiation testing requires the ability to re-fabricate previously irradiated fuel rods within INL's Hot Fuel Examination Facility (HFEF) due to the high radiological doses of the irradiated fuel. The scope aims to establish basic re-fabrication capabilities at INL to enable these follow-on irradiation experiments. Basic re-fabrication consists of inspection to determine the desired segment to be cut from the source material, sectioning or cutting the segment, de-fueling the ends of the segment to make space for the instrumentation and new rodlet endcaps to be welded in place, removal of the oxide layer from the ends to prepare the surface for welding new endcaps, welding the new endcaps in place, evacuating the rodlet and backfilling with the desired gas mixture to the desired pressure, seal welding the rodlet, and inspections. Enabling these re-fabrication tasks requires designing, prototyping, and testing the Rodlet End Welding System to perform these functions prior to final equipment being approved for installation in the hot cell (5d window). Application of limited instrumentation to the rodlet will be possible in this system.

For advanced rod measurement designs, a complementary reinstrumentation system is under development in collaboration with Halden that will provide a modular and flexible platform for performing precision operations and handling of delicate instrumentation. The state-of-the-art Halden reinstrumentation process entails neutron or x-ray radiography, cutting and drilling out the fuel at the ends of the rod, removing the oxide layer from cladding surfaces, cryogenic freezing (e.g. filling the rod with liquid CO_2 and freeze in liquid N_2), drilling a rod center hole for instrumentation, outgassing in vacuum, insertion of centerline thermocouples, welding rod end plugs, adding inert gas and seal welding, and conducting helium leak checks. This system will allow INL to add instrumentation to irradiated fuel rods and improve future testing.

Accomplishments

The most critical fuel performance data needs are typically associated with time dependent evolution and end-of-life fuel properties and behavior. However, the in-situ instrumentation required to collect that data is rarely robust enough to function effectively through the entire life cycle of a test. As such, the ability to extract fuel from the reactor and install instrumentation at the critical testing points is an essential capability. This capability (including techniques and infrastructure) was first developed at the INL in support of transient and steady state testing but was shelved in the 1980's when US investment in advanced light water reactor (LWR) technology waned. International partners like Halden subsequently adopted these tools and built robust research and development (R&D) missions around them. The revival of US interest in advanced LWR technology (e.g., burnup extension and ATF) coupled with the closure of international facilities (like Halden and OSIRIS) calls for the recovery of these capabilities to support domestic priorities.

The basic re-fabrication system or Rodlet Endcap Welding System (REWS), (see Figure 1) has completed detailed requirements and design in coordination with Automated Welding Systems (AMET) and INL. The REWS will have programmable controls, a purge enclosure for decreased porosity, cross seam servo with approximately seventeen inch travel, high frequency or touch start arc initiation capabilities, and through hole headstock to enable longer rodlets with diameters between one-quarter inch and one-half inch. AMET is also fabricating the In Cell Weld Under Pressure System (ICWUPS) (Figure 2). The ICWUPS will have programmable controls, the ability to pressurize rodlets from 2 Mpa (290 psi) to 15.5 Mpa (2250 psi) with inert atmosphere, wand ill have horizontal rod loading with vertical weld position. These systems will be installed in HFEF in the decontamination cell at window 5D. Both systems are anticipated to be delivered and setup in North Holmes Laboratory (NHL) the end of September 2020 to for mockup testing.

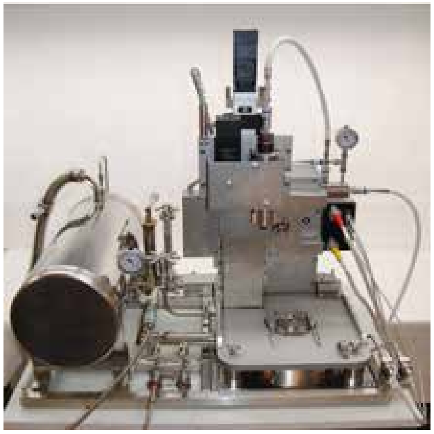


Figure 1. Rodlet End Welding System (REWS).

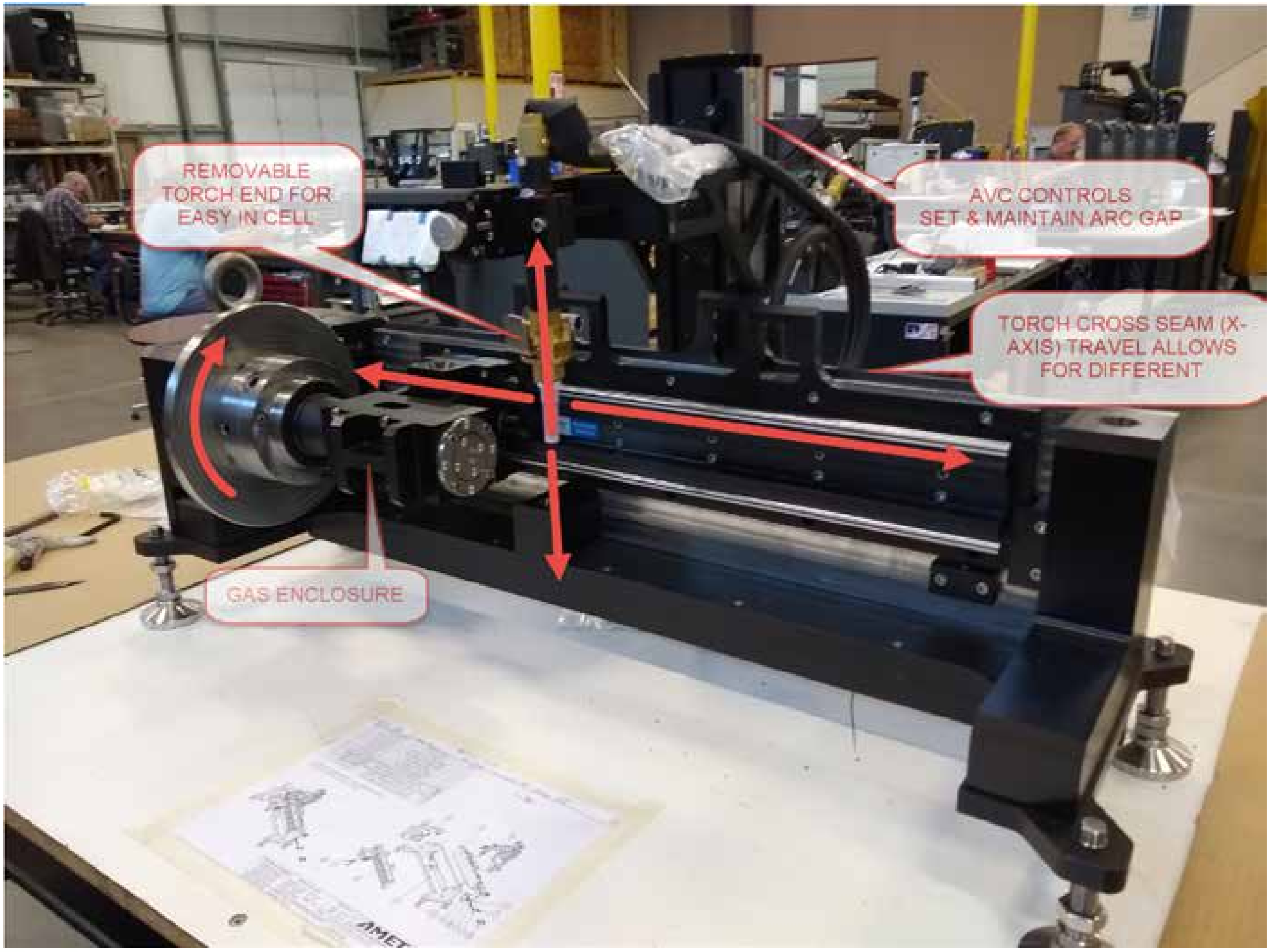


Figure 2. In Cell Weld Under Pressure System (ICWUPS).

The advanced instrumentation program is being informed by AFC and executed by NE-5 Advanced Sensors & Instrumentation (ASI) program. To develop the advanced instrumentation for irradiated fuel rods INL has been working with Institute for Energy Technology (IFE) to derive the detailed requirements and design for a prototype re-instrumentation system (cryo-drilling system). A purchase order has been placed to purchase a re-instrumentation system from IFE with an expected delivery in December 2020. This system will take approximately three months to install in HTTL once received. A Halden like welding system is also planned for delivery in April 2021, followed by three months of setup in HTTL. Both systems will be installed to support development of techniques and ultimately the final design for the system planned for the TREAT Experiment Support Building (TESB).

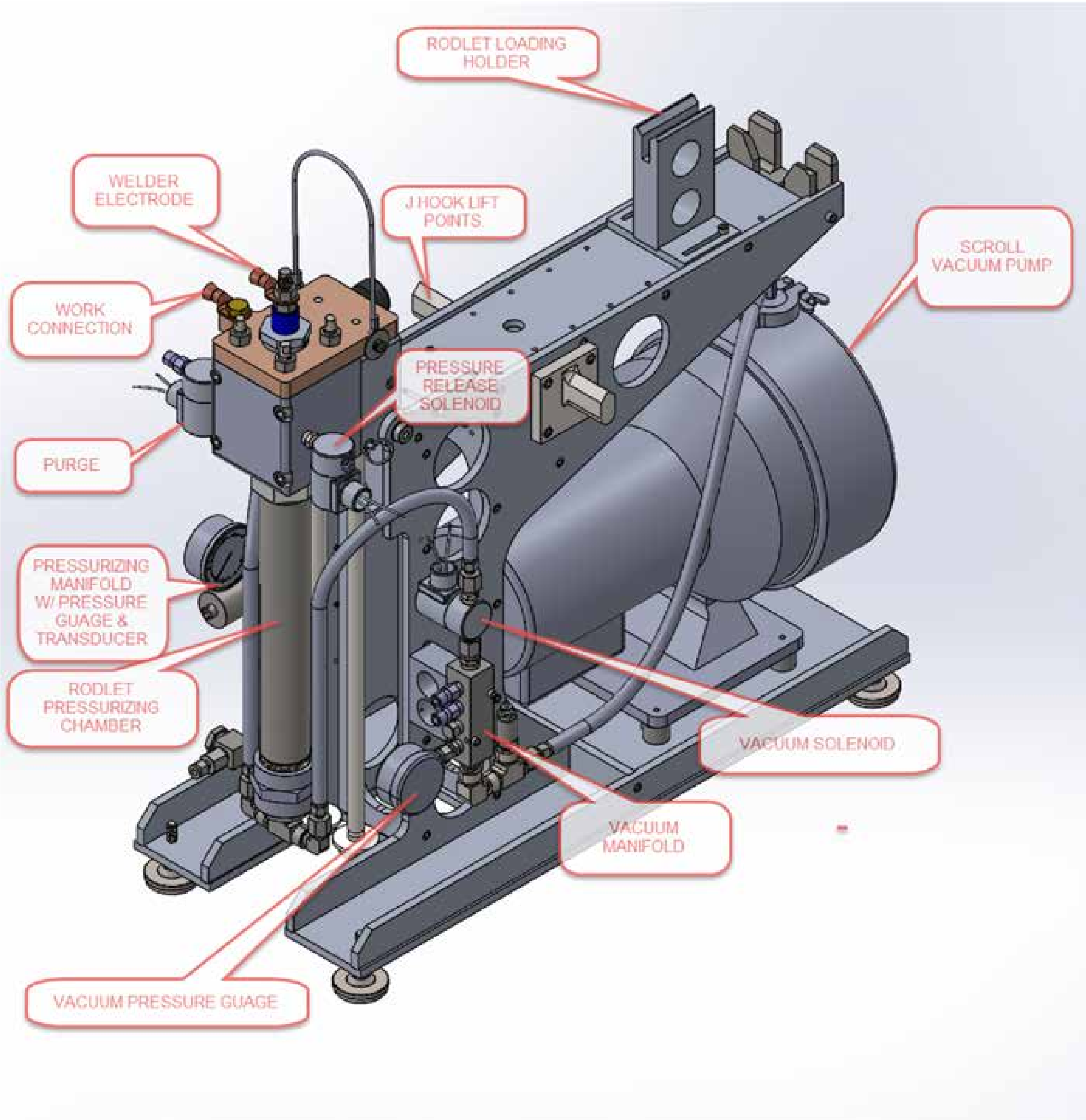


Figure 3. Drilling Unit.

4.2 I-LOOP PROJECT

Expand Access and Capability to the Advanced Test Reactor (ATR) Medium I with a Flowing Water Loop

Principal Investigator: Thomas Maddock
Team Members/ Collaborators: Nate Oldham

This research allows advanced fuel qualification and a pressurized water loop in the ATR reactor.

The I-loop project expands access and capability to the Advanced Test Reactor (ATR) medium I with a flowing water loop, allowing testing of fuels and materials in prototypic conditions.

Project Description

The closure of the Halden Boiling Water Reactor (HBWR) and growing demand for extending light water reactor (LWR) fuel performance limits, has created urgency to find a testing solution to support near-term fuel testing needs. This need comes at a time when industry, via Electric Power Research Institute (EPRI), and the Department of Energy (DOE), via the Accident Tolerant Fuels (ATF) Program, are pushing for performance and capability increases for new fuels. The I-loop project aims to fill the gap created by HBWR's closure and provide a similar capability at ATR, by installing a flowing water loop in one medium-I position and providing access to the other medium-I positions for future experiments. In order to provide better access to the I-positions of ATR, a new Top Head Closure Plate (Mark II) has been designed, expanding the capabilities of the ATR for continued fuel and materials testing. To support the flowing water loop the refurbishment of the 1A cubicle will be required including new pumps, piping, heat exchangers, and other equipment. In order to provide the neutronics to approximate prototypic LWR nuclear grade Zr-2.5NB will be used for the experiment tubes. The improvements will enable advanced fuel qualification to continue at the Idaho National Laboratory (INL), and support the industry and DOE desire for these capabilities in the United States, to support the continued safe use of existing LWR reactors, and provide for future fuel testing for next generation reactors.

Accomplishments

The program went through multiple funding determinations to determine its status as a capital asset project. Necessary paperwork, project execution plan, schedule and budgets were developed to support this determination.

APPENDIX

5.1 PUBLICATIONS

Author	Title	Publication
S. Agarwal et al.	Revealing irradiation damage along with the entire damage range in ion-irradiated SiC/SiC composites using Raman spectroscopy	Journal of Nuclear Materials 526 (2019): 151778
A.F. Ali, J.P. Gorton, N.R. Brown, K.A. Terrani, C.B. Jensen, Y. Lee and E.D. Blandford	Surface wettability and pool boiling Critical Heat Flux of Accident Tolerant Fuel cladding-FeCrAl alloys	Nucl. Eng. Des. 338 (2018): 218–231 doi:10.1016/ j.nucengdes.2018.08.024.
A. Ali, H.-G. Kim, K. Hattar, S. Briggs, D. Jun Park, J. Hwan Park, Y. Lee	Ion irradiation effects on Cr-coated zircaloy-4 surface wettability and pool boiling critical heat flux	Nucl. Eng. Des. 362 (2020): 110581 doi:10.1016/ j.nucengdes.2020.110581.
J.L. Baker, G.Wang, T. L. Ulrich, J.T. White, E.R. Batista, P. Yang, R.C. Roback, C. Park and H. Xu	High-Pressure Structural Behavior and Elastic Properties of U_3Si_5: A Combined Synchrotron XRD and DFT Study	Journal of Nuclear Materials (2020) doi:10.1016/j.jnucmat.2020.152373.
Beausoleil et al.	Integrating Advanced Modeling and Accelerated Testing for a Modernized Fuel Qualification Paradigm	Nuclear Technology
N.R. Brown, B.E. Garrison, R.R Lowden, M.N. Cinbiz and K.D. Linton	Mechanical failure of fresh nuclear grade iron–chromium–aluminum (FeCrAl) cladding under simulated hot zero power reactivity initiated accident conditions	Journal of Nuclear Materials (2020): 152352
C.H. Bumgardner, F.M. Heim, D.C. Roache, A. Jarama, P. Xu, R. Lu, E. J. Lahoda, B.P. Croom, C.P. Deck and X. Li	Unveiling hermetic failure of ceramic tubes by digital image correlation and acoustic emission	Journal of the American Ceramic Society (2019)
J.R. Burns, R. Hernandez, K.A. Terrani, A.T. Nelson and N.R. Brown	Reactor and fuel cycle performance of light water reactor fuel with 235U enrichments above 5%.	Annals of Nuclear Energy, 142 (2020): 107423
N. Capps, R. Sweet, B.D. Wirth, Andrew Nelson, K.A. Terrani	Development and demonstration of a methodology to evaluate high burnup fuel susceptibility to pulverization under a loss of coolant transient	Nuclear Engineering and Design 366 (2020): 110744, ISSN 0029-5493.
N. Capps, Y. Yan, A. Raftery, Z. Burns, T. Smith, K.A. Terrani, K. Yueh, M. Bales and K.D. Linton	Integral LOCA fragmentation test on high-burnup fuel	Nuclear Eng. and Design 367 (2020): 110811 https://doi.org/10.1016/j.nucengdes.2020.110811

Author	Title	Publication
L. Capriotti and J. M. Harp	Characterization of a minor actinides bearing metallic fuel pin irradiated in EBR-II	Journal of Nuclear Materials 539 (2020): 152279
H. J. M. Chichester, B. A. Hilton, S. L. Hayes, L. Capriotti, P. G. Medvedev and D. L. Porter	Irradiation Performance of Nonfertile (Pu-MA-Zr) Fast Reactor Metal Fuels	Journal of Nuclear Materials: 152480, in press.
Y. Cui, E. Aydogan, J. G. Gigax, Y. Wang, A. Misra, S. A. Maloy and N. Li	In situ micropillar compression to examin radiation-induced hardeninig mechanishsms of FeCrAl alloys	Acta Materialia, submitted.
T. Dabney, G. Johnson, H. Yeom, B. Maier, J. Walters and K. Sridharan	Experimental Evaluation of Cold Spray FeCrAl Alloys Coated Zirconium-alloy for Potential Accident Tolerant Fuel Cladding	Nuclear Materials and Energy 21 (2019): 100715
P. Deng, M. Karadge, R. B. Rebak, V. K. Gupta, B. C. Prorok and X. Lou	The origin and formation of oxygen inclusions in austenitic stainless steels manufactured by laser powder fusion	Additive Manufacturing 35 (2020): 101334
P.J. Doyle et al.	Evaluation of the effects of neutron irradiation on first-generation corrosion mitigation coatings on SiC for accident-tolerant fuel cladding	Journal of Nuclear Materials (2020): 152203
P.J. Doyle et al.	The effects of neutron and ionizing irradiation on the aqueous corrosion of SiC	Journal of Nuclear Materials (2020): 152190
P.J. Doyle, S. Zinkle and S. S. Raiman	Hydrothermal corrosion behavior of CVD SiC in high temperature water	Journal of Nuclear Materials (2020): 152241
B.P. Eftink, M.E. Quintana, T.J. Romero, C.Xu, D.T. Hoelzer, T.A. Saleh and S.A. Maloy	Shear Punch Testing of Neutron-Irradiated HT-9 and 14YWT	JOM 72 (2020)
L.J. Evitts, S.C. Middleburgh, E. Kardoulaki, I. Ipatova, M.J.D. Rushton and W.E. Lee	Influence of boron isotope ratio on the thermal conductivity of uranium diboride (UB_2) and zirconium diboride (ZrB_2)	Journal of Nuclear Materials (2020): 1–7 https://doi.org/10.1016/j.jnucmat.2019.151892.
J. Gigax, A. Torrez, Q. McCulloch, H. Kim, N. Li and S. Maloy	Sizing up mechanical testing: Comparison of microscale and mesoscale mechanical testing techniques on a FeCrAl welded tube	J. Mater. Res. (2020): doi: 10.1557/jmr.2020.195

Author	Title	Publication
B. Gong, C. Lu, P. Lei, K. E. Metzger, E. J. Lahoda, F.A. Boylan, K. Yang, J. Fay, J.Harp and J. Lian	Cr Doped U_3Si_2 Composite Fuels under Steam Corrosion	Corrosion Science (2020), under review.
B. Gong, T. Yao, P. Lei, C. Lu, K. E. Metzger, E. J. Lahoda, F. A. Boylan, A. Mohamad, J. Harp, A. T. Nelson and J. Lian	U_3Si_2 and UO_2 composites densified by spark plasma sintering for accident-tolerant fuels	Journal of Nuclear Materials 534 (2020): 152147
J.P. Gorton, S.K. Lee, Y. Lee and N.R. Brown	Comparison of experimental and simulated critical heat flux tests with various cladding alloys: Sensitivity of iron-chromium-aluminum (FeCrAl) to heat transfer coefficients and material properties	Nucl. Eng. Des. 353 (2019): 110295 doi.org/10.1016/j.nucengdes.2019.110295
J. M. Harp, L. Capriotti, D. L. Porter and J. I. Cole	U-10Zr and U-5Fs: Fuel/cladding chemical interaction behavior differences	528 (2020) 151840
M. He and Y. Lee	Application of machine learning for prediction of critical heat flux: Support vector machine for data-driven CHF look-up table construction based on sparingly distributed training data points	Nucl. Eng. Des. 338 (2018): 189–198 doi:10.1016/j.nucengdes.2018.08.005.
M. He and Y. Lee	Revisiting heater size sensitive pool boiling critical heat flux using neural network modeling: Heater length of the half of the Rayleigh-Taylor Instability Wavelength maximizes CHF	Therm. Sci. Eng. Prog. 14 (2019): 100421 doi:10.1016/j.tsep.2019.100421.
M. He and Y. Lee	Application of Deep Belief Network for Critical Heat Flux Prediction on Microstructure Surfaces	Nuclear Technology 206 (2020): 358–374 doi:10.1080/00295450.2019.1626177.
F. M. Heim, J.T. Daspit, O. B. Holzmond, B. P. Croom and X. Li	Analysis of tow architecture variability in biaxially braided composite tubes	Composites Part B: Engineering 190 (2020): 107938
F. M. Heim, J.T. Daspit and X.Li	Quantifying the effect of unit cell irregularity on the performance of biaxially braided composite tubes	Composites Part B: Engineering, accepted

Author	Title	Publication
A. Jena, A. Kossolapov, B. Phillips and M. Bucci	The wettability of next-generation nuclear fuel cladding materials in a steam-saturated high-pressure environment	In preparation
K.E. Johnson, D. L. Adorno, V. Kocevski, T. L. Ulrich, J. T. White, A. Claisse, J.W. McMurrary and T. M. Besmann	Impact of Fission Product Inclusion on Phase Development in U_3Si_2 Fuel	Journal of Nuclear Materials 537 (2020): 152235
H. Jo, H. Yeom, E. Gutierrez, K. Sridharan and M. Corradini	Evaluation of Critical Heat Flux of ATF Candidate Coating Materials in Pool Boiling	Nuclear Engineering and Design 354 (2019): 110166
K. A. Kane, S.K. Lee, S.B. Bell, N.R. Brown and B.A. Pint	Burst behavior of nuclear grade FeCrAl and Zircaloy-2 fuel cladding under simulated cyclic dryout conditions	Journal of Nuclear Materials 539 (2020): 152256
E. Kardoulaki, J.T. White, D.D. Byler, D.M. Frazer, A.P. Shivprasad, T.A. Saleh, B. Gong, T. Yao, J. Lian and K.J. Mcclellan	Thermophysical and mechanical property assessment of UB_2 and UB_4 sintered via spark plasma sintering	J. Alloys Compd. 818 (2020) 1–14 https://doi.org/10.1016/j.jallcom.2019.153216.
V. Kocevski, D. A. Lopes, A. J. Claisse and T. M. Besmann	Understanding the interface interaction between U_3Si_2 fuel and SiC cladding	Nature communications 11 (1) (2020): 1-8
T. Koyanagi, Y. Katoh and T. Nozawa	Design and strategy for next-generation silicon carbide composites for nuclear energy	Journal of Nuclear Materials (2020): 152375
A.G. Le Coq, R.N. Morris, C.M. Petrie and J.R. Burns	Post Irradiation Examination Results of Miniature Fuel Specimens Irradiated in the High Flux Isotope Reactor	Transactions of the American Nuclear Society 121 (2019): 615-618
D. Lee, B. Elward, P. Brooks, R. Umretiya, J. Rojas, M. Bucci, R. B. Rebak and M. Anderson	Enhanced flow boiling heat transfer on chromium coated zircaloy-4 using cold spray technique for Accident Tolerant Fuel (ATF) materials	Applied Thermal Engineering, under review
S.K. Lee, Y. Lee, N.R. Brown and K.A. Terrani	Elucidating the Impact of Flow on Material-Sensitive Critical Heat Flux and Boiling Heat Transfer Coefficients: An Experimental Study with Various Materials	International J. Heat Mass Transf. 158 (2020): 119970 doi:10.1016/j.ijheatmasstransfer.2020.119970.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Author	Title	Publication
S.K. Lee, M. Liu, N.R. Brown, K.A. Terrani, E.D. Blandford, H. Ban, C.B. Jensen and Y. Lee	Comparison of steady and transient flow boiling critical heat flux for FeCrAl accident tolerant fuel cladding alloy, Zircaloy, and Inconel	Int. J. Heat Mass Transf. 132 (2019): 643–654 doi:10.1016/j.ijheatmasstransfer.2018.11.141.
S.K. Lee, M. Liu, N.R. Brown and K.A. Terrani, Y. Lee	Effect of Heater Material and Thickness on the Steady-State Flow Boiling Critical Heat Flux	Nuclear Technology 206 (2020): 339–346 doi:10.1080/00295450.2019.1670010.
A.S. Losko, L. Daemen, P. Hosemann, H. Nakotte, A. Tremsin, S.C. Vogel, P. Wang and F.H. Wittman	Separation of Uptake of Water and Ions in Porous Materials Using Energy Resolved Neutron Imaging	JOM (2020): 1-8
Q. McCulloch, J. Gigax, and P. Hosemann	Femtosecond laser ablation for mesoscale specimen evaluation	JOM 72(4) (2020): 1694
C. Mckinney, T.J. Gerczak, and J. Harp	Sample Preparation for 3D Characterization of Irradiated Fuel	United States: N. p., 2020. Web. doi:10.2172/1631238
P.A. Mouche et al.	Characterization of PVD Cr, CrN, and TiN coatings on SiC	Journal of Nuclear Materials 527 (2019): 151781
P.A. Mouche and K. A. Terrani	Steam pressure and velocity effects on high temperature silicon carbide oxidation	Journal of the American Ceramic Society 103.3 (2020): 2062-2075
N.E. Peterson, D. Malta, S.C. Vogel, B. Clausen, S. Jana, V.V. Joshi and S.R. Agnew	The role of ternary alloying elements in eutectoid transformation of U–10Mo alloy part II. In and ex-situ neutron diffraction-based assessment of eutectoid phase transformation kinetics in U-9.8 Mo-0.2 X alloy (X= Cr, Ni or Co)	Journal of Nuclear Materials 540 (2020):152383
C.M. Petrie, A. Le Coq, D. Richardson, C. Hobbs, G. Helmreich, J. Burns and J. Harp	Monolithic ATF MiniFuel Sample Capsules Ready for HFIR Insertion	United States: N. p., 2020. Web. doi:10.2172/1615787
S. Raiman et al.	Hydrothermal corrosion of 2nd generation FeCrAl alloys for accident tolerant fuel cladding	Journal of Nuclear Materials (2020): 152221
S. S. Raiman, K. G. Field, R. B. Rebak, Y. Yamamoto and K. A. Terrani	Hydrothermal corrosion of 2nd generation FeCrAl alloys for accident tolerant fuel cladding	Journal of Nuclear Materials 536 https://doi.org/10.1016/j.jnucmat.2020.152221.

Author	Title	Publication
R. B. Rebak, L. Yin and P. L. Andresen	Resistance of ferritic FeCrAl alloys to stress corrosion cracking for light water reactor fuel cladding applications	Corrosion Journal, NACE International, https://doi.org/10.5006/3632
B. Reed, R. Wang, R. Y. Lu and J. Qu	Autoclave Grid-to-Rod Fretting Wear Evaluation of a Candidate Cladding Coating for Accident-Tolerant Fuel	Wear (2020) (submitted)
J. Schulthess*, N. Woolstenhulme, A. Craft, J. Kane, N. Boulton, W. Chuirazzi, A. Winston, A. Smolinski, C. Jensen, D. Kamerman and D. Wachs	Non-Destructive Post irradiation Examination Results of the First Modern Fueled Experiments in TREAT	Journal of Nuclear Materials 541 (2020): 152442 https://doi.org/10.1016/j.jnucmat.2020.152442
G. Y. Su, C. Wang, L. Zhang, J. H. Seong, B. Phillips, R. Kommayosula and M. Bucci	Investigation of flow boiling heat transfer and boiling crisis on a rough surface using infrared thermometry	International Journal of Heat and Mass Transfer 160 (2020): 120134
K.A. Terrani, B.C. Jolly and J.M. Harp	Uranium nitride tristructural-isotropic fuel particle	Journal of Nuclear Materials 531 (2020): 152034 doi:10.1016/J.JNUCMAT.2020.152034.
T.L. Ulrich, S.C. Vogel, D.A. Lopes, V. Kocevski, J.T. White, E.S. Sooby and T.M. Besmann	Phase stability of U_5Si_4, USi, and U_2Si_3 in the uranium–silicon system	Journal of Nuclear Materials 540 (2020): 152353
T.L. Ulrich, S.C. Vogel, J.T. White, D.A. Andersson, E.S. Wood and T.M. Besmann	High temperature neutron diffraction investigation of U_3Si_2	Materialia 9 (2020):100580
R. V. Umretiya, B. Elward, D. Lee, M. Anderson, R. B. Rebak and J. V. Rojas	Mechanical and chemical properties of PVD and cold spray Cr-coatings on Zircaloy-4	Journal of Nuclear Materials 541 (2020): 152420. https://doi.org/10.1016/j.jnucmat.2020.152420.
R. V. Umretiya, S. Vargas, D. Galeano, R. Mohammadi, C.E. Castano and J.V. Rojas	Effect of surface characteristics and environmental aging on wetting of Cr-coated Zircaloy-4 accident tolerant fuel cladding material	Journal of Nuclear Materials (2020): 152163.https://doi.org/10.1016/j.jnucmat.2020.152163
S.C. Vogel, M.A. Bourke, A.E. Craft, J.M. Harp, C.T. Kelsey, J. Lin, A.M. Long, A.S. Losko, P. Hosemann, K.J. McClellan and M. Roth	Advanced Postirradiation Characterization of Nuclear Fuels Using Pulsed Neutrons	JOM 72(1) (2020): 187-196

Author	Title	Publication
S.C. Vogel, J.C. Fernandez, D.C. Gautier, N. Mitura, M. Roth and K.F. Schoenberg	Short-Pulse Laser-Driven Moderated Neutron Source	EPJ Web of Conferences 231 (2020): 01008). EDP Sciences.
W.J. Williams, M.A. Okuniewski, S.C. Vogel and J. Zhang	In Situ Neutron Diffraction Study of Crystallographic Evolution and Thermal Expansion Coefficients in U-22.5 at.% Zr During Annealing	JOM (2020): 1-9
E. Sooby Wood, C. Moczygemba, G. Robles, Z. Acosta, B.A. Brigham, C.J. Grote, K.E. Metzger and L. Cai	High temperature steam oxidation dynamics of U_3Si_2 with alloying additions: Al, Cr, and Y	Journal of Nuclear Materials 533 (2020)
N. Woolstenhulme, A. Fleming, T. Holschuh, C. Jensen, D. Kamerman and D. Wachs	Core-to-Specimen Energy Coupling Results of the First Modern Fueled Experiments in TREAT	Annals of Nuclear Energy (2020)
N. Woolstenhulme, C. Jensen, C. Folsom, R. Armstrong, J. Yoo and D. Wachs	Thermal-hydraulic and Engineering Evaluations of New LOCA Testing Methods in TREAT	Nuclear Technology (2020), accepted Manuscript
T. Yao, B. Gong, P. Lei, C. Lu, P. Xu, E. Lahoda and J. Lian	UO_2+x-5 vol.% ZrB_2 Composite Fuels with Boron Retention and Enhanced Oxidation Resistance	Ceramic International (2020), in press
H. Yeom, E. Gutierrez, H. Jo, K. Sridharan and M. Corradini	High Temperature Water Corrosion and Pool Boiling Critical Heat Flux of Accident Tolerant Fuel Cladding Materials	Nuclear Engineering and Design (2020), under review

5.2 FY-20 LEVEL 2 MILESTONES

Work Package Title	Site	Work Package Manager	Level 2 Milestone
Work Package Title	Site	Work Package Manager	Milestone Title
ATF Transient Testing - INL	INL	Emerson, Leigh Ann	Perform transients on ATF concepts in SETH capsule (~3 transients)
Irradiation of ATF Concepts in HFIR - ORNL	ORNL	Petrie, Christian	Monolithic ATF MiniFuel Sample Capsules Ready for HFIR Insertion
ATF-2 ATR Loop Design - INL	INL	Hoggard, Gary	New ATF-2 Tier 5/6 ready for new Westinghouse rodlets and ATR insertion
Evolution of Fuel Structure, Chemistry, & Properties following Irradiation - ORNL	ORNL	Harp, Jason	Issue Report Summarizing Complete PIE of Gen1 MiniFuel Samples
LOCA Testing - INL	INL	Dempsey, Doug	Complete preliminary design of TREAT LOCA test
Irradiation Testing in ATR - INL	INL	Murdock, Chris	FAST-1 accelerated test ready for insertion in ATR
ATF Transient Testing - INL	INL	Emerson, Leigh Ann	Complete MARCH-SERTTA commissioning tests
Remote Handling Capability for Transient Testing - INL	INL	Bybee, Micheal	Demonstrate ability to load an ATF-2 rodlet into a TREAT capsule in the HFEF hot cell
Re-fabrication and Instrumentation Techniques - INL	INL	Cole, Mark	Demonstrate ability to perform re-fabrication of fuel rods in mockup
AFC Coordination and Integration - INL	INL	Mai, Edward	Complete Draft 2020 Accomplishments Report

5.3 AFC NUCLEAR ENERGY UNIVERSITY PROJECTS (NEUP) GRANTS

Active Projects Awarded in 2015

Nuclear Energy University Cooperative Agreements

Lead University	Title	Principal Investigator
Massachusetts Institute of Technology	Multilayer Composite Fuel Cladding for LWR Performance Enhancement and Severe Accident Tolerance	Michael Short

Active Projects Awarded in 2016

Nuclear Energy University Cooperative Agreements

Lead University	Title	Principal Investigator
University South Carolina	Phase Equilibria and Thermochemistry of Advanced Fuels: Modeling Burnup Behavior	Theodore Besmann
North Carolina State University	Microstructure Experiments-Enabled MARMOT Simulations of SiC/SiC-based Accident Tolerant Nuclear Fuel System	Jacob Eapen
Purdue University	Microstructure, Thermal, and Mechanical Properties Relationships in U and UZr Alloys	Maria Okuniewski
Pennsylvania State University	A Coupled Experimental and Simulation Approach to Investigate the Impact of Grain Growth, Amorphization, and Grain Subdivision in Accident Tolerant U_3Si_2 Light Water Reactor Fuel	William Walters
University of Idaho	A Science Based Approach for Selecting Dopants in FCCI-Resistant Metallic Fuel Systems	Indrajit Charit
The Ohio State University	Alloying Agents to Stabilize Lanthanides Against Fuel Cladding Chemical Interaction: Tellurium and Antimony Studies	Christopher Taylor

Active Projects Awarded in 2017

Nuclear Energy University Cooperative Agreements

Lead University	Title	Principal Investigator
University of Wisconsin-Madison	Extreme Performance High Entropy Alloys (HEAs) Cladding for Fast Reactor Applications	Adrien Couet
University of Wisconsin-Madison	Critical Heat Flux Studies for Innovative Accident Tolerant Fuel Cladding Surfaces	Michael Corradini
Colorado School of Mines	Development of Advanced High-Cr Ferritic/ Martensitic Steels	Kester Clarke
Massachusetts Institute of Technology	Determination of Critical Heat Flux and Leidenfrost Temperature on Candidate Accident Tolerant Fuel Materials	Matteo Bucci
University of New Mexico	An Experimental and Analytical Investigation into Critical Heat Flux (CHF) Implications for Accident Tolerant Fuel (ATF) Concepts	Anil Prinja
Missouri University of Science and Technology	Gamma-ray Computed and Emission Tomography for Pool-Side Fuel Characterization	Joseph Graham
Virginia Commonwealth University	Evaluation of Accident Tolerant Fuels Surface Characteristics in Critical Heat Flux Performance	Jessika Rojas
University of New Mexico	Nanostructured Composite Alloys for Extreme Environments	Osman Anderoglu

Active Projects Awarded in 2018

Nuclear Energy University Cooperative Agreements

Lead University	Title	Principal Investigator
University of California, Berkeley	Understanding of degradation of SiC/SiC materials in nuclear systems and development of mitigation strategies	Peter Hosemann
University of Minnesota, Twin Cities	Probabilistic Failure Criterion of SiC/SiC Composites Under Multi-Axial Loading	Jialiang Le
University of Wisconsin-Madison	Advanced Coating and Surface Modification Tech- nologies for SiC-SiC Composite for Hydrothermal Corrosion Protection in LWR	Kumar Sridharan
University of Michigan	Mechanistic Understanding of Radiolytically Assisted Hydrothermal Corrosion of SiC in LWR Coolant Environments	Gary Was
University of Florida	Multiaxial Failure Envelopes and Uncertainty Quantification of Nuclear-Grade SiCf/SiC Woven Ceramic Matrix Tubular Composites	Ghatu Subhash
University of Notre Dame	Radiolytic Dissolution Rate of Silicon Carbide	David Bartels
University of South Carolina	Development of Multi-Axial Failure Criteria for Nuclear Grade SiCf-SiCm Composites	Xinyu Huang
University of California, Berkeley	Bridging the length scales on mechanical property evaluation	Peter Hosemann
Purdue University	Microstructure-Based Benchmarking for Nano/ Microscale Tension and Ductility Testing of Irradiated Steels	Janelle Wharry
University of Utah	Benchmarking Microscale Ductility Measurements	Owen Kingstedt
University of Nebraska, Lincoln	Bridging microscale to macroscale mechanical property measurements and predication of performance limitation for FeCrAl alloys under extreme reactor applications	Jian Wang
Virginia Polytechnic Institute and State University	C-SiOC-SiC Coated Particle Fuels for Advanced Nuclear Reactors	Kathy Lu
University of Tennessee at Knoxville	A novel and flexible approach for converting LWR UNF fuel into forms that can be used to fuel a variety of Gen-IV reactors	Craig Barnes

Active Projects Awarded in 2019

Nuclear Energy University Cooperative Agreements

Lead University	Title	Principal Investigator
University of Pittsburgh	Thermal Conductivity Measurement of Irradiated Metallic Fuel Using TREAT	Heng Ban
The Ohio State University	Neutron Radiation Effect on Diffusion between Zr (and Zircaloy) and Cr for Accurate Lifetime Prediction of ATF	Ji-Cheng Zhao
North Carolina State University	Novel miniature creep tester for virgin and neutron irradiated clad alloys with benchmarked multiscale modeling and simulations	Korukonda Murty
University of South Carolina	Remote laser based nondestructive evaluation for post irradiation examination of ATF cladding	Lingyu Yu
University of Tennessee at Knoxville	Radiation-Induced Swelling in Advanced Nuclear Fuel	Maik Lang
University of Minnesota, Twin Cities	High throughput assessment of creep behavior of advanced nuclear reactor structural alloys by nano/microindentation	Nathan Mara

Active Projects Awarded in 2020

Nuclear Energy University Cooperative Agreements

Lead University	Title	Principal Investigator
University of Wisconsin-Madison	Investigation of Degradation Mechanisms of Cr-coated Zirconium Alloy Cladding in Reactivity Initiated Accidents (RIA)	Hwasung Yeom
University of Wisconsin-Madison	Maintaining and building upon the Halden legacy of In-situ diagnostics	Michael Corradini
University of California, Berkeley	Femtosecond Laser Ablation Machining & Examination - Center for Active Materials Processing (FLAME-CAMP)	Peter Hosemann
Rensselaer Polytechnic Institute	Chemical Interaction and Compatibility of Uranium Nitride with Liquid Pb and Alumina-forming Austenitic Alloys	Jie Lian
Georgia Institute of Technology	Linear and nonlinear guided ultrasonic waves to characterize cladding of accident tolerant fuel (ATF)	Laurence Jacobs

5.5 ACRONYMS

AEC .... Atomic Energy Commission

AFC .... Advanced Fuels Campaign

AFIR .... Autoclave Fretting Rig

AFM .... Advanced Fuel Management

AFQ .... Accelerated Fuel Qualification

AL .... Air Liquide

AMET .... Automated Welding Systems

APMT .... Advanced Powder Metallurgy Tubing

ARCTIC .... Advanced Reactor Concepts In Thermal-spectrum Capability

ARES .... Advanced Reactor Experiments for Sodium Fast Reactor Fuels

ARL .... Army Research Lab

ASI .... Advanced Sensors & Instrumentation

ATF .... Accident Tolerant Fuel

ATR .... Advanced Test Reactor

BDBA .... Beyond Design Basis Accident

BNL .... Brookhaven National Laboratory

BOC .... Beginning of Cycle

BOL .... Beginning of Life

BWR .... Boiling Water Reactor

CEA .... Commissariat à l' Énergie Atomique

CHF .... Critical heat flux

CIC .... Core Internal Change-out

CMR .... Chemistry and Metallurgy Research

CRADA .... Cooperative Research and Development Agreement

CS .... Cold Spray

CS .... Conventional Sintering

CTE .... Coefficient of Thermal Expansion

CVD .... Chemical Vapor Deposition

DBA .... Design Basis Accident

DCD .... Design Control Document

DCMS .... DC Magnetron Sputtering

DFT .... Density Functional Theory

DISECT .... Disc Irradiation for Separate Effects Testing with Control of Temperature

DOE .... Department of Energy

DOT .... Department of Transportation

dpa .... Displacements per Atom

E & S .... Evaluation and Screening

EATF .... Enhanced Accident Tolerant Fuel

EBR .... Experimental Breeder Reactor

ECF .... Energy Coupling Factor

EDC .... Expansion Due to Compression

EDM .... Electro-Discharge Machining

EDS .... Energy Dispersive X-Ray Spectroscopy

EM2 .... Energy Multiplier Module

EOC .... End of Cycle

EOL .... End of Life

EPRI .... Electric Power Research Institute

ERNI .... Energy-Resolved Neutron Imaging

FAST .... Field Assisted Sintering Technique

FAST .... Fission Accelerated Steady-state Testing

FAST-OA .... Fission Accelerated Steady-state Testing – Outer A position

FAST-SI .... Fission Accelerated Steady-state Testing – Small I position

FCCI .... Fuel Cladding Chemical Interaction

FFTF .... Fast Flux Test Facility

FFRD .... Fuel Fragmentation, Relocation, and Dispersal

FGR .... Fission Gas Release

FIB .... Focused Ion Beam

FIMA .... Fission of Initial Metal Atoms

F/M .... Ferritic-Martensitic

FOM .... Figure of Merit

FRL .... Fuels Research Laboratory

FS .... Flash Sintering

FY .... Fiscal Year

GA .... General Atomics

GE .... General Electric

GET .... Gamma Emission Tomography

GFY .... Government Fiscal Year

GNF .... Global Nuclear Fuels

GTRF .... Grid-to-Rod-Fretting

GWd/MTU .... Gigawatt-days Per Metric Ton of Uranium

HALEU .... High Assay Low-Enriched Uranium

HBFF .... High Burnup Fuel Fragmentation

HBHE .... High Burnup - Higher Enrichment

HBS .... High Burnup Structures

HBWR .... Halden Boiling Water Reactor

HFEF .... Hot Fuel Examination Facility

HFIR .... High Flux Isotope Reactor

HiPIMS .... Hybrid High Power Impulse Magnetron Sputtering

HIPPO .... High Pressure/Preferred Orientation

HTC-NB .... Heat Transfer Coefficient of Nucleate Boiling

IAC .... Industry Advisory Committee

ICWUPS .... In Cell Weld Under Pressure System

IFE .... Institute for Energy Technology

INL .... Idaho National Laboratory

IR........Infrared

IRT........Integral Recycling Test

JAEA........Japanese Atomic Energy Agency

KAERI........Korean Atomic Energy Research Institute

LAMDA........Low Activation Materials Development & Analysis

LANL........Los Alamos National Laboratory

LANSCE........Los Alamos Neutron Science Center

LDRD........Laboratory Directed Research and Development

LFA........Laser Flash Analysis

LHGR........Linear Heat Generation Rates

LOCA........Loss of Coolant Accident

LRM........Linear Reactivity Model

LTA........Lead Test Assemblies

LTR........Lead Test Rod

LVDT........Linear Variable Differential Transformer

LWR........Light Water Reactor

MARCH........Minimal Activation Retrievable Capsule Holder

MARCH-SERTTA........Minimal Activation Retrievable Capsule Holder–Static Environment Rodlet Transient Test Apparatus

MCNP5........Monte Carlo N-Particle 5

MFC........Materials and Fuels Complex

MITR........Massachusetts Institute of Technology Reactor

MMU........Manchester Metropolitan University

MOX........Mixed Oxide

MSC........Master Sintering Curve

nCT........Neutron Computed Tomography

NE........Nuclear Energy

NEA........Nuclear Energy Agency

NEAMS........Nuclear Energy Advanced Modeling and Simulation

NEI	Nuclear Energy Institute
NEUP	Nuclear Energy University Project
NFA	Nanostructured Ferritic Alloys
NFD	Nippon Nuclear Fuel Development
NHL	North Holmes Laboratory
NI	Nanoidentation
NNL	National Nuclear Laboratory
NRAD	Neutron Radiography Reactor
NRC	Nuclear Regulatory Commission
NSUF	Nuclear Science User Facilities
NTRD	Nuclear Technology Research & Development
ODS	Oxide Dispersion Strengthened
ONB	Onset of Nucleate Boiling
ONE	Office of Nuclear Energy
ORNL	Oak Ridge National Laboratory
OSU	Oregon State University
PBF	Power Burst Facility
PCI	Pellet Clad Interactions
PCMI	Pellet-clad Mechanical Interaction
PCS	Primary Coolant System
PI	Principal Investigator
PIE	Postirradiation Examination
PIRT	Phenomena Identification and Ranking Tables
PNNL	Pacific Northwest National Laboratory
PVD	Physical Vapor Deposited
PVD	Physical Vapor Deposition
PWR	Pressurized Water Reactor
R&D	Research & Development
RAC	Radial, Axial and Capsule

RD&D .... Research, Development, and Demonstration

REWS .... Rodlet Endcap Welding System

RIA .... Reactivity-Initiated Accident

RPI .... Rensselaer Polytechnic Institute

RT .... Room Temperature

SATS .... Severe Accident Test Station

SEM .... Scanning Electron Microscopy

SERTTA .... Static Environment Rodlet Transient Test Apparatus

SETH .... Separate Effects Test Holder

SFR .... Sodium Fast Reactor

SiC .... Silicon Carbide

SPERT .... Special Power Excursion Reactor Test

STEM .... Scanning Transmission Electron Microscopy

TD .... Theoretical Density

TE .... Total Elongation

TEM .... Transmission Electron Microscopy

TESB .... TREAT Experiment Support Building

THOR .... Temperature Heat sink Overpower Response

TIG .... Tungsten Inert Gas

Tm .... Top Middle

TM .... Temperature Monitor

TMS .... The Mineral, Metal and Materials Society

TREAT .... Transient Reactor Test Facility

TRISO .... Tri-structural Isotropic

TWERL .... TREAT Water Environment Recirculating Loop

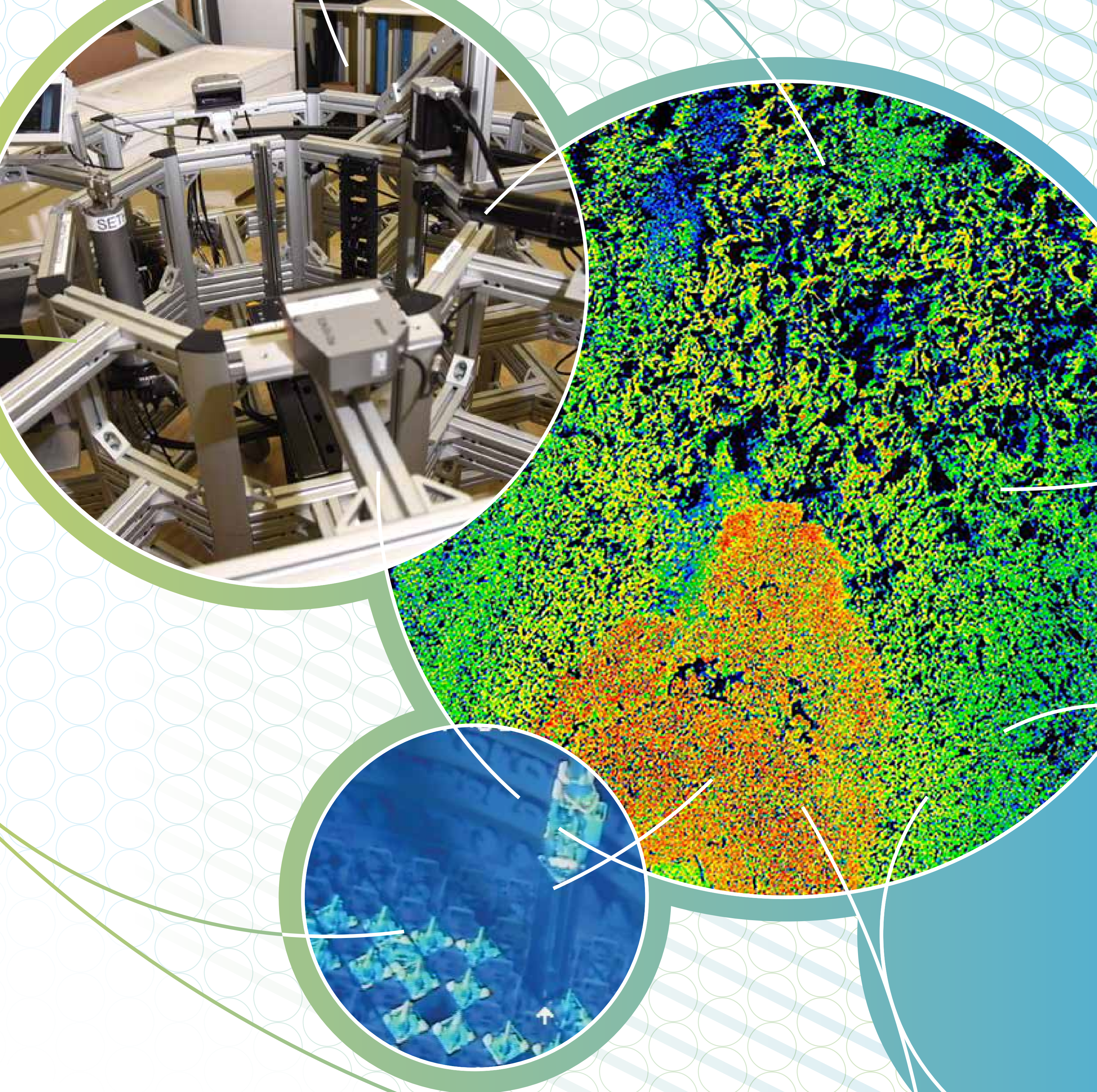
UC .... Uranium Carbide

UE .... Uniform Elongation

U.S. .... United States

UN .... Uranium Nitride

UNM	University of New Mexico
UofI	University of Idaho
USC	University of South Carolina
UTA	University of Texas-Austin
UTK	University of Tennessee-Knoxville
UTS	Ultimate Tensile Strength
UW	University of Wisconsin
VCU	Virgina Commonwealth University
VTR	Versatile Test Reactor
WALT	Westinghouse Advanced Test Loop
WEC	Westinghouse Electric Company
XCT	X-ray Computed Tomography
XRD	X-Ray Diffraction
YS	Yield Strength
YWT	Alloys Containing Y, W and Ti



Advanced Fuels Campaign